U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)
Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. The fuel system consists of: arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. The Mechanical Engineering Branch reviews the design of control rod drive mechanisms in SRP Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Coolability, in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident.

Fuel failure criteria and coolability criteria that involve thermal-hydraulic considerations are provided by the Core Performance Branch to the Analysis Branch for implementation in SRP Section 4.4. The Analysis Branch provides hydraulic loads under SRP Section 4.4 to the Core Performance Branch for evaluation (in SRP Section 4.2) of fuel assembly mechanical response under normal and accident conditions. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Analysis Branch for use in estimating the radiological consequences of plant releases.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

The fuel system review covers the following specific areas.

A. Design Bases

The principles and related assumptions of the fuel system design should be reviewed. These bases may be expressed as explicit numbers or as general criteria. The bases will include traditional fuel design limits, industry codes and standards, and limits related to the safety analysis (i.e., related to fuel damage, rod failure, or coolability requirements). Once such limits are approved in the safety evaluation report, they become the specified acceptable fuel design limits referred to in General Design Criterion 10 (Ref. 1). The design bases should reflect the safety review objectives as described above.

B. Description and Design Drawings

The fuel system description and design drawings are reviewed. In general, the description will emphasize product specifications rather than process specifications.

C. Design Evaluation

The performance of the fuel system during normal operation, anticipitated operational occurrences, and postulated accidents is reviewed to determine if all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. The review consists of an evaluation of operating experience, direct experimental comparisons, detailed mathematical analyses, and other information.

D. Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel is performed by the licensee to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. On-line fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to ensure against reactivity loss. The testing, inspection, and surveillance plans along with their reporting provisions are reviewed by CPB to ensure that the important fuel design considerations have been addressed.

II. ACCEPTANCE CRITERIA

A. Design Bases

The fuel system design bases must reflect the four objectives described in Subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following:

1. Fuel System Damage

Fuel system damage includes fuel rod failure, which is discussed below in Subsection II-A-2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Such damage criteria should include the following to be complete.

(a) Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that ar. obtained by methods similar to those given in Section III of the ASME Code (Ref. 2) are acceptable. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles (Ref. 3). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in paragraph (a) above should be limited. The allowable fretting wear should be stated in the safety analysis report and the stress and fatigue limits in paragraphs (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the safety analysis report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed by the Analysis Branch as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are provided by the Analysis Branch as described in SRP Section 4.4.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B_4C) are used.

2. Fuel Rod Failure

Fuel rod failure is defined as the loss of fuel rod hermeticity. Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to known failure mechanisms during normal operation and anticipated operational occurrences. Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. A fuel failure criterion should be given for each known failure mechanism. Such criteria should address the following to be complete.

(a) Overheating: No useful mechanistic criteria exist at present for fuel rod failure due to overheating. However, to show that overheating will be avoided, it will be sufficient to show that (1) cladding temperatures do not greatly exceed the coolant temperature and (2) fuel melting does not occur.

Adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nucleate boiling (DNB) or boiling transition condition in the core is satisfied. The review of this criterion is detailed in SRP Section 4.4.

For a severe reactivity initiated accident (RIA), Regulatory Guide 1.77 (Ref. 4) relies on a DNB criterion for determining failures in PWRs, whereas a radial average energy density of 170 cal/g is accepted for BWRs under zero and low power conditions. Other limits may be more accurate for an RIA, but continued approval of these limits may be given until generic studies yield improvements.

Although a DNB criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. Although there is at present little experience with other approaches, positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

The second criterion used to assure that the cladding does not overheat is that fuel melting will not occur. There would otherwise be concern that molten fuel might contact the cladding and cause local hotspots.

This criterion also avoids the axial relocation of molten fuel that could cause local overheating.

(b) Pellet/Cladding Interaction (PCI): There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures: (1) the uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten core to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (a) to ensure that overheating of the cladding would not occur.

(c) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 5) for UO_2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources. For other materials clad in Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 mg H_2O per cm^3 of hot void volume within the Zircaloy cladding has been shown (Ref. 6) to be insufficient for primary hydride formation.

(d) Cladding Collapse: If axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

(e) Bursting: Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. Although fuel suppliers may use different rupture-temperature vs differential-pressure curves, an acceptable curve should be similar to the one determined by Oak Ridge National Laboratory (Ref. 7). This criterion is included in the ECCS evaluation model required by Appendix K (Ref. 8).

(f) Mechanical Fracturing: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be

assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified.

(g) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an abnormal operational occurrence or a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of Subsection II-A-1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during transients.

3. Fuel Coolability

Coolability has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. Coolability criteria should include the following to be complete:

(a) Cladding Embrittlement: Oxygen contamination and hydriding in Zircaloy cladding are the primary causes of cladding embrittlement. For the LOCA, Appendix K addresses these phenomena with a criterion of 2200°F peak cladding temperature and a criterion of 17% maximum cladding oxidation. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 1800°F previously described in Reference 9 is no longer needed in CP and OL reviews.) Specific temperature and oxidation criteria have not been derived for other accidents, but should they be needed, Appendix K can be used as guidance.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents (RIAs), such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. Observing the 280 cal/g limit specified by Regulatory Guide 1.77 prevents widespread fragmentation and dispersal of the fuel and avoids generating pressure pulses in the primary system during an RIA. This 280 cal/g limit should be used for PWRs and BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry.

Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Structural Deformation: It is sufficient, but not necessary, to show that no permanent deformation of spacer grids, channel boxes, fuel rods, guide tubes, and other structural members results from mechanical loads. This can be accomplished by demonstrating that the expected load is less than the grid crushing strength or that the load produces a maximum stress that is less than the material's irradiated yield stress at the appropriate temperature.

Some structural deformation may not always be avoidable. In those cases, the degree of deformation must be determined before establishing coolability. For the LOCA, structural deformation should not cause the 2200°F cladding temperature and 17% cladding oxidation limits to be exceeded. Specific criteria have not been derived for other accidents, but, should they be needed, Appendix K should be used as guidance.

A Branch Technical Position is under development to quantify margins and other aspects of structural deformation criteria. Reference 10 will also provide some guidance to the reviewer on this matter.

(e) Fuel Rod Ballooning: For the LOCA analysis, Appendix K requires that flow blockage resulting from cladding ballooning (swelling) be taken into account in the analysis of core flow distribution. Flow blockage models must be based on applicable data (Refs. 7, 11, and 12) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (e) of Subsection II-A-2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated. The flow blockage model evaluation is provided to the Analysis Branch for incorporation in the comprehensive ECCS model evaluation to show that the 2200°F cladding temperature and 17% cladding oxidation limits are not exceeded. The reviewer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

B. Description and Design Drawings

The reviewer should see that the fuel system description and design drawings are complete enough to provide an accurate representation and to supply information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

Type and metallurgical state of the cladding
Cladding outside diameter
Cladding inside diameter
Cladding i.d. roughness
Pellet outside diameter
Pellet roughness
Pellet density
Pellet resintering data
Pellet length
Pellet dish dimensions
Burnable poison content
Insulator pellet parameters
Fuel column length
Overall rod length
Rod internal void volume
Fill gas type and pressure
Sorbed gas composition and content
Spring and plug dimension
Fissile enrichment
Equivalent hydraulic diameter
Coolant pressure

The following design drawings have also been found necessary for an acceptable fuel system description:

Fuel assembly cross section
Fuel assembly outline
Fuel rod schematic
Spacer grid cross section
Guide tube and nozzle joint
Control rod assembly cross section
Control rod assembly outline
Control rod schematic
Burnable poison rod assembly cross section
Burnable poison rod assembly outline
Burnable poison rod schematic
Orifice and source assembly outline

C. Design Evaluation

The methods of demonstrating that the design bases are met must be reviewed. Those methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in PSARs and FSARs by reference.

1. Operating Experience

Operating experience with fuel systems of the same or similar design should be described. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience needed not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed when practical to determine the characteristics of

the new design. No definitive requirements have been developed regarding those design features that must be tested prior to irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

Spacer grid structural tests
Control rod structural and performance tests
Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The following phenomena that have been tested in this manner in new designs will serve as a guide to the reviewer:

Fuel and burnable poison rod growth
Fuel rod bowing
Fuel assembly growth
Fuel assembly bowing
Channel box wear and distortion
Fuel rod ridging (PCI)
Crud formation
Fuel rod integrity
Holddown spring relaxation
Spacer grid spring relaxation
Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished prior to operation of a full core of that design. This inability to perform in-reactor testing may result from an incompatability of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see Subsection II-D below).

3. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Phenomenological models that should be reviewed include the following:

Radial power distribution
Fuel and cladding temperature distribution
Burnup distribution in the fuel
Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers

Densification of the fuel
Thermal expansion of the fuel and cladding
Fission gas production and release
Solid and gaseous fission product swelling
Fuel restructuring and relocation
Fuel and cladding dimensional changes
Fuel-to-cladding heat transfer coefficient
Thermal conductivity of the gas mixture
Thermal conductivity in the Knudsen domain
Fuel-to-cladding contact pressure
Heat capacity of the fuel and cladding
Growth and creep of the cladding
Rod internal gas pressure and composition
Sorption of helium and other fill gases
Cladding oxide and crud layer thickness
Cladding-to-coolant heat transfer coefficient*

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (Refs. 13 and 14). Examples of previous fuel performance code reviews are given in References 15 through 18.

(b) Densification Effects: In addition to its effect on fuel temperatures (discussed above), densification affects (1) core power distributions (power spiking, see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR, see SRP Section 4.4), and (3) the potential for cladding collapse. Densification magnitudes for power spike and LHGR analyses are discussed in Reference 19 and in Regulatory Guide 1.126 (Ref. 20). Models for cladding collapse times must also be reviewed, and previous review examples are given in References 21 and 22.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference 23. Interim methods that may be used prior to compliance with this guidance are given in Reference 24. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: The mechanical response of fuel assemblies to impact loads should be analyzed with a multi-assembly spring/mass representation. Because of the complexity and related uncertainties of these analyses, independent generic audit calculations should be performed to compare analytical methods with the NRC audit code. An example of such a review and audit calculation is given in Reference 25.

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by CPB. The models are empirical and should be compared with relevant data. Examples of such data and a previous review are contained in References 7, 11, 12, and 26.

*Although needed in fuel performance codes, this model is reviewed by the Analysis Branch as described in SRP Section 4.4.

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II-A-1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27) as required by Appendix K. For non-LOCA applications, other correlations may be used if justified.

(h) Fission Product Inventory: The available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by assumptions in Regulatory Guides (Refs. 4, 28-30). These assumptions should be used until improved calculational methods are approved by CPB (see Ref. 31).

D. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

1. Testing and Inspection of New Fuel

Testing and inspection plans for new fuel should include verification of cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Details of the manufacturer's testing and inspection programs should be documented in quality control reports, which should be referenced and summarized in the safety analysis report. The program for on-site inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. Where the overall testing and inspection programs are essentially the same as for previously approved plants, a statement to that effect should be made. In that case, the details of the programs need not be included in the safety analysis report, but an appropriate reference should be cited and a (tabular) summary should be presented.

2. On-line Fuel System Monitoring

The applicant's on-line fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. Reference 32 evaluates several common detection methods and should be utilized in this review.

Surveillance is also needed to assure that B_4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a

cladding defect. Periodic reactivity worth tests such as described in Reference 33 are acceptable.

3. Post-irradiation Surveillance

A post-irradiation fuel surveillance program should be described for each plant to detect anomalies or confirm expected fuel performance. The extent of an acceptable program will depend on the history of the fuel design being considered, i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features.

For a fuel design like that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, or crud deposition. There should also be a commitment in the program to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel.

In addition to the plant-specific surveillance program, there should exist a continuing fuel surveillance effort for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program.

For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with prototype testing discussed in Subsection II-C-2. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.

III. REVIEW PROCEDURES

For construction permit (CP) applications, the review should assure that the design bases set forth in the preliminary safety analysis report (PSAR) meet the acceptance criteria given in Subsection II-A. The CP review should further determine from a study of the preliminary fuel system design that there is reasonable assurance that the final fuel system design will meet the design bases. This judgment may be based on experience with similar designs.

For operating license (OL) applications, the review should confirm that the design bases set forth in the final safety analysis report (FSAR) meet the acceptance criteria given in Subsection II-A and that the final fuel system design meets the design bases.

Much of the fuel system review is generic and is not repeated for each similar plant. That is, the reviewer will have reviewed the fuel design or certain aspects of the fuel design in previous PSARs, FSARs, and licensing topical reports. All previous reviews on which the current review is dependent should be referenced so that a completely documented safety evaluation is contained in the plant safety evaluation report. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Certain generic reviews have also been performed by CPB reviewers with findings issued as NUREG- or WASH-series reports. At the present time these reports include References 9, 19, 31, 32, 34 and 35, and they should all be appropriately cited in the plant safety evaluation report. Applicable Regulatory Guides (Refs. 4, 20, 28-30) and Branch Technical Positions (there are none at present) should also be mentioned in the plant safety evaluation reports. Deviation from these guides or positions should be explained. After briefly discussing related previous reviews, the plant safety evaluation should concentrate on areas where the application is not identical to previously reviewed and approved applications and areas related to newly discovered problems.

Analytical predictions discussed in Subsection II-C-3 will be reviewed in PSARs, FSARs, or licensing topical reports. When the methods are being reviewed, calculations by the staff may be performed to verify the adequacy of the analytical methods. Thereafter, audit calculations will not usually be performed to check the results of an approved method that has been submitted in a safety analysis report. Calculations, benchmarking exercises, and additional reviews of generic methods may be undertaken, however, at any time the clear need arises to reconfirm the adequacy of the method.

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information has been provided to satisfy the requirements of this SRP Section and that the evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The fuel system of the _____ plant has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents.

"The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions.

"The applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated.

"The applicant has also provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has made a commitment to perform on-line fuel failure monitoring and post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

"On the basis of our review of the fuel system design, we conclude that the applicant has met all the requirements of the applicable regulations, current regulatory positions, and good engineering practice."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. "Rules for Construction of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, Section III, 1977.

3. W. J. O'Donnel and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nucl. Sci. Eng. 20, 1 (1964).

4. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

5. "Standard Specification for Sintered Uranium Dioxide Pellets," ASTM Standard C776-76, Part 45, 1977.

6. K. Joon, "Primary Hydride Failure of Zircaloy-Clad Fuel Rods," Trans. Am. Nucl. Soc. 15, 186 (1972).

7. R. H. Chapman, "Multirod Burst Test Program Quartarly Progress Report for April - June 1977," Oak Ridge National Laboratory Report ORNL/NUREG/TM-135, December 1977.

8. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

9. "Technical Report on Densification of Light Water Reactor Fuels," AEC Regulatory Staff Report WASH-1236, November 14, 1972.

10. "Safety Evaluation Report Related to Operation of North Anna Power Station," USNRC Report NUREG-0053, Supp. 7, August 1977.

11. F. Erbacher, "Single and Multirod Tests, Transient and Steady State, Internal Conduction Heating," Fifth NRC Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, November 11, 1977.

12. R. H. Chapman, "Some Preliminary Results of Single Rod and Multirod Tests With Internal Heaters," NRC Zircaloy Cladding Review Group Meeting, Silver Spring, Maryland, January 18, 1978.

13. C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko and L. J. Parchen, "User's Guide for GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratory Report BNWL-1897, November 1975.

14. C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko and L. J. Parchen, "GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratory Report BNWL-1898, November 1975.

15. R. H. Stoudt, D. T. Buchanan, B. J. Buescher, L. L. Losh, H. W. Wilson and P. J. Henningson, "TACO - Fuel Pin Performance Analysis, Revision 1," Babcock & Wilcox Report BAW-10087A, Rev. 1, August 1977.

16. "Fuel Evaluation Model," Combustion Engineering Report CENPD-139-A, July 1974 (Approved version transmitted to NRC April 25, 1975).

17. "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," AEC Regulatory Staff Report, December 14, 1973.

18. "Technical Report on Densification of Exxon Nuclear PWR Fuels," AEC Regulatory Staff Report, February 27, 1975.

19. R.O. Meyer, "The Analysis of Fuel Densification," USNRC Report NUREG-0085, July 1976.

20. Regulatory Guide 1.126, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification."

21. Memorandum from V. Stello, NRC, to R.C. DeYoung, Subject: Evaluation of Westinghouse Report, WCAP-8377, Revised Clad Flattening Model, dated January 14, 1975.

22. Memorandum from D. F. Ross, NRC, to R. C. DeYoung, Subject: CEPAN -- Method of Analyzing Creep Collapse of Oval Cladding, dated February 5, 1976.

23. Memorandum from D. F. Ross, NRC, to D. B. Vassallo, Subject: Request for Revised Rod Bowing Topical Reports, dated May 30, 1978.

24. Memorandum from D. F. Ross and D. G. Eisenhut, NRC, to D. B. Vassallo and K. R. Goller, Subject: Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors, dated February 16, 1977.

25. R. L. Grubb, "PWR Fuel Assembly Mechanical Response Analysis," Idaho National Engineering Laboratory Report RE-S-76-164, September 1976; Amendment 1, RE-E-77-103, January 1977.

26. Letter from D. F. Ross, NRC, to A.E. Scherer, Combustion Engineering, dated March 22, 1978.

27. L. Baker and L. C. Just, "Studies of Metal-Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium - Water Reaction," Argonne National Laboratory Report ANL-6548, May 1962.

28. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors."

29. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

30. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

31. "The Role of Fission Gas Release in Reactor Licensing," USNRC Report NUREG-75/077, November 1975.

32. B. L. Siegel and H. H. Hagen, "Fuel Failure Detection in Operating Reactors," USNRC Report NUREG-0401, March 1978.

33. "Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2," USNRC Report NUREG-0308, Supp. 2 (to be issued).

34. B. L. Siegel, "Evaluation of the Behavior of Waterlogged Fuel Rod Failures in LWRs," USNRC Report NUREG-0303, March 1978.

35. R. O. Meyer, C. E. Beyer and J. C. Voglewede, "Fission Gas Release from Fuel at High Burnup," USNRC Report NUREG-0418, March 1978.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.3 NUCLEAR DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The review of the nuclear design of the fuel assemblies, control systems, and reactor core is carried out to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and to assure conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28.

The review of the nuclear design under this SRP section, the review of the fuel system design under SRP Section 4.2, the review of the thermal and hydraulic design under SRP Section 4.4, and the review of the transient and accident analyses under the SRP section for Chapter 15 of the applicant's safety analysis report (SAR), are all necessary in order to confirm that the requirements defined above are met.

The specific areas of interest in the nuclear design include:

1. Confirmation that design bases are established as required by the appropriate General Design Criteria.

2. The areas concerning core power distribution. These are:

   a. The presentation of expected or possible power distributions including normal and extreme cases for steady state and allowed load-follow transients and covering a full range of reactor conditions of time in cycle, allowed control rod positions, and possible fuel burnup distributions. The power distributions should include power spikes from fuel densification.

   b. The presentation of the core power distributions as axial, radial, and local distributions and peaking factors to be used in the transient and accident analyses.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

c. The translation of the design power distributions into operating power distributions, including instrument-calculation correlations, operating procedures and measurements, and necessary limits on these operations.

d. The requirements for instruments, the calibration and calculations involved in their use, and the uncertainties involved in translation of instrument readings into power distributions.

e. Limits and setpoints for actions, alarms, or scram for the instrument systems and demonstration that these systems can maintain the reactor within design power distribution limits.

f. Measurements in previous reactors and critical experiments and their use in the uncertainty analyses, and measurements to be made on the reactor under review, including startup confirmatory tests and periodically required measurements.

g. The translation of design limits, uncertainties, operating limits, instrument requirements, and setpoints into technical specifications.

3. The areas concerning reactivity coefficients. These are:

a. The applicant's presentation of calculated nominal values for the reactivity coefficients such as the moderator coefficient, which involves primarily effects from density changes and takes the form of temperature, void, or density coefficients; the Doppler coefficient; and power coefficients. The range of reactor states to be covered includes the entire operating range from cold shutdown through full power, and the extremes reached in transient and accident analyses. It includes the extremes of time in cycle and an appropriate range of control rod insertions for the reactor states.

b. The applicant's presentation of uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude by examination of the accuracy of the methods used in calculations (SAR Section 4.3.3), and comparison where possible with reactor experiments.

c. The applicant's combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady state analysis (primarily control requirements, SAR Section 4.3.2.4), stability analyses (SAR Section 4.3.2.8), and the transient and accident analyses presented in SAR Chapter 15.

4. The areas concerning reactivity control requirements and control provisions. These are:

a. The control requirements and provisions for control necessary to compensate for long-term reactivity changes of the core. These reactivity changes occur because of depletion of the fissile material in the fuel, depletion of burnable poison in some of the fuel rods, and buildup of fission products and transuranium isotopes.

b. The control requirements and provisions for control needed to compensate for the reactivity change caused by changing the temperature of the reactor from the hot, zero power condition to the cold shutdown condition.

c. The control requirements and provisions for control needed to compensate for the reactivity effects caused by changing the reactor power level from full power to zero power.

d. The control requirements and provisions for control needed to compensate for the effects on the power distribution of the high cross-section Xe-135 isotope.

e. The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation.

f. The applicant's analysis and experimental basis for determining the reactivity worth of a "stuck" control rod of highest worth.

g. The provision of two independent control systems.

5. The areas of control rod patterns and reactivity worths. These are:

a. Descriptions and figures indicating the control rod patterns expected to be used throughout a fuel cycle. This includes operation of single rods or of groups or banks of rods, rod withdrawal order, and insertion limits as a function of power and core life.

b. Descriptions of allowable deviations from the patterns indicated above, such as for misaligned rods, stuck rods, or rod positions used for spatial power shaping.

c. Descriptions, tables, and figures of the maximum worths of individual rods or banks as a function of position for power and cycle life conditions appropriate to rod withdrawal transients and rod ejection or drop accidents. Descriptions and curves of maximum rates of reactivity increase associated with rod withdrawals, experimental confirmation of rod worths or other factors justifying the reactivity increase rates used in control rod accident analyses, and equipment, administrative procedures, and alarms which may be employed to restrict potential rod worths should be included.

d. Descriptions and graphs of scram reactivity as a function of time after scram initiation and other pertinent parameters, including methods for calculating the scram reactivity.

6. The area of criticality of fuel assemblies. Discussions and tables giving values of K_{eff} for single assemblies and groups of adjacent fuel assemblies up to the number required for criticality, assuming the assemblies are dry and also immersed in water, are reviewed.

7. The areas concerning analytical methods. These are:

   a. Descriptions of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, burnup, and stability.

   b. The data base used for neutron cross-sections and other nuclear parameters.

   c. Verification of the analytical methods by comparison with measured data.

8. The areas concerning pressure vessel irradiation. These are:

   a. Neutron flux spectrum above 1 MeV in the core, at the core boundaries, and at the inside pressure vessel wall.

   b. Assumptions used in the calculations; these include the power level, the use factor, the type of fuel cycle considered, and the design life of the vessel.

   c. Computer codes used in the analysis.

   d. The data base for fast neutron cross sections.

   e. The geometric modeling of the reactor, support barrel, water annulus, and pressure vessel.

   f. Uncertainties in the calculation.

The AB reviews the adequacy of limits on power distribution during normal operation in connection with their analyses of the thermal-hydraulic design, under SRP Section 4.4.

The ICSB, under SRP Section 7, reviews the adequacy of proposed instrumentation to meet the requirements for maintaining the reactor operating state within defined limits.

II. ACCEPTANCE CRITERIA

1. The basic acceptance criteria in the area of nuclear design are the General Design Criteria (GDC) related to the reactor core and reactivity control systems (Ref. 1).

   a. GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

   b. GDC 11 requires that in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

   c. GDC 12 requires that power oscillations which could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

   d. GDC 13 requires provision of instrumentation and controls to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain them within prescribed operating ranges.

   e. GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions.

   f. GDC 25 requires that no single malfunction of the reactivity control system (this does not include rod ejection or dropout) causes violation of the acceptable fuel design limits.

   g. GDC 26 requires that two independent reactivity control systems of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

   h. GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

   i. GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

2. The following discussions present less formal criteria and guidelines used in the review of the nuclear design.

a. There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady state or load-following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limits (SAR Section 4.4), scram limits (SAR Chapter 7) and transient and accident analyses (SAR Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 2200°F peak cladding temperature allowed for loss-of-coolant accidents (LOCA), are not exceeded during design basis accidents. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed.

The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that:

(1) A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods.

(2) A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).

Criteria for acceptable values and uses of uncertainties in operation, instrumentation numerical requirements, limit settings for alarms or scram, frequency

and extent of power distribution measurements, and use of excore and incore instruments and related correlations and limits for offsets and tilts, all vary with reactor type. They can be found in staff safety evaluation reports and in appropriate sections of the technical specifications and accompanying bases for reactors similar to the reactor under review (Ref. 2). The CPB has enunciated a Branch Technical Position CPB 4.3-1 for Westinghouse reactors which employ constant axial offset control (Ref. 7).

Acceptance criteria for power spike models can be found in staff technical reports on fuel densification (Ref. 3).

Generally, special or newly emphasized problems related to core power distributions will not be a direct part of normal reviews but will be handled in special generic reviews. Fuel densification effects and the related power spiking and the use of uncertainties in design limits are examples of these areas.

b. The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states "...the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity," and is considered to be satisfied in light water reactors by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient such as may exist in pressurized water reactors at beginning of core life.

The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgment to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations; experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.

c. Acceptance criteria relative to control rod patterns and reactivity worths include:

(1) The predicted control rod worths and reactivity insertion rates must be reasonable bounds to values that may occur in the reactor. These values are used in the transient and accident analyses and judgment as to the

adequacy of the uncertainty allowances are made in the review of the transient and accident analyses.

(2) Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a CPB position to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.

d. There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and data base should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate and encompass a sufficient range.

III. REVIEW PROCEDURES

The review procedures below apply in general to both the construction permit (CP) and operating license (OL) stage reviews. At the CP stage, parameter values and certain design aspects may be preliminary and subject to change. At the OL stage, final values of parameters should be used in the analyses presented in the SAR. The review of the nuclear design of a plant is based on the information provided by the applicant in the safety analysis report, as amended, and in meetings and discussions with the applicant and his contractors and consultants. This review in some cases will be supplemented by independent calculations performed by the staff or staff consultants. Files of audit calculations are maintained for reference by the reviewer (Ref. 5).

1. The reviewer confirms, as part of the review of specific areas of the nuclear design outlined below, that the design bases, design features, and design limits are established in conformance with the GDCs listed in Section II.

2. The reviewer examines the information presented in the SAR to determine that the core power distributions for the reactor can reasonably be expected to fall within the design limits throughout all normal (steady state and load-follow) operations, and that the instrument systems employed, along with the information processing systems and alarms, will reasonably assure the maintenance of the distributions within these limits for normal operation.

For a normal review, many areas related to core power distribution will have been examined in generic reviews or earlier reviews of reactors with generally similar core characteristics and instrument systems. A large part of the review on a particular case may then involve comparisons with information from previous application reviews. The comparisons may involve the shapes and peaking factors of normal and limiting distributions over the range of operating states of the reactor, the effects of power spikes from densification, assigned uncertainties and their use, calculation methods and data used, correlations used in control processes, instrumentation requirements, information processing methods including computer use, setpoints

for operational limits and alarm limits, and alarm limits for abnormalities such as flux asymmetries.

An important part of this review, at the OL stage, covers the relevant sections of the proposed technical specifications, where power distributions and related controls such as control rod limits are discussed. Here the instrument requirements, limit settings, and measurement frequencies and requirements are set forth in full detail. The comparison of technical specifications should reveal any differences between essentially identical reactors or any lack of difference between reactors with changed core characteristics. Where these occur, the reviewer must assess the significance and validity of the differences or lack of differences.

This review and comparison may be supplemented with examinations of related topical reports from reactor vendors, generic studies by staff consultants, and startup reports from operating reactors which contain information on measured power distributions (Ref. 4).

3. The reviewer determines from the applicant's presentations that suitably conservative reactivity coefficients have been developed for use in reactor analyses such as those for control requirements, stability, and transients and accidents. The reviewer examines:

   a. The applicability and accuracy of methods used for calculations including the use of more accurate check calculations such as the use of Monte Carlo techniques for Doppler models.

   b. The models involved in the calculations such as the model used for effective fuel temperature in Doppler coefficient analyses.

   c. The reactor state conditions assumed in determining values of the coefficients. For example, the pressurized water reactor (PWR) moderator temperature coefficient to be used in the steam line break analysis is usually based on the reactor condition at end of life with all control rods inserted except the most reactive rod, and the moderator temperature in the hot standby range.

   d. The applicability and accuracy of experimental data from critical experiments and operating reactors used to determine or justify uncertainty allowances. Measurements during startup and during the cycle of moderator temperature coefficients and full power Doppler coefficients in the case of PWRs, and results of measurements of transients during startup in the case of boiling water reactors (BWRs), should be examined. As part of the review, comparisons are made between the values and uncertainty allowances for reactivity coefficients for the reactor under review and those for similar reactors previously reviewed and approved. Generally, many essential areas will have been covered during earlier reviews of similar reactors. The reviewer notes any differences

in results for essentially identical reactors and any lack of differences for reactors with changed core characteristics, and judges the significance and validity of any differences or lack of differences.

4. The review procedures in the area of reactivity control requirements and control provisions are as follows:

   a. The reviewer determines that two independent reactivity control systems of different design are provided.

   The reviewer examines the tabulation of control requirements, the associated uncertainties, and the capability of the control systems, and determines by inspection and study of the analyses and experimental data that the values are realistic and conservative.

   c. The reviewer determines that one of the control systems is capable of returning the reactor to the cold shutdown condition and maintaining it in this condition, at any time in the cycle. It is necessary that proper allowance be made for all of the mechanisms that change the reactivity of the core as the reactor is taken from the cold shutdown state to the hot, full power operating state. The reviewer should determine that proper allowance is made for the decrease in fuel temperature, moderator temperature, and the loss of voids (in BWRs) as the reactor goes from the power operating range to cold shutdown.

   d. The reviewer determines that one of the control systems is capable of rapidly returning the reactor to the hot standby (shutdown) condition from any power level at any time in the cycle. This requirement is met by rapid insertion of control rods in all current light water reactors. Proper allowance for the highest worth control rod being stuck in the full-out position must be made. In PWRs, operational reactivity control is carried out by movement of control rods and by adjustments of the concentration of soluble poison in the coolant. The reviewer must pay particular attention to the proposed rod insertion limits in the power operating range, to assure that the control rods are capable of rapidly reducing the power and maintaining the reactor in the hot standby condition. This is an important point because the soluble poison concentration in the coolant could be decreased in order to raise reactor power, while the control rods were left inserted so far that in the event of a scram (rapid insertion of control rods), the available reactivity worth of the control rods on full insertion would not be enough to shut the reactor down to the hot standby condition.

   e. The reviewer determines that each of the independent reactivity control systems is capable of controlling the reactivity changes resulting from planned, normal power operation. This determination is made by comparing the rate of reactivity change resulting from planned, normal operation to the capabilities of each of

the two control systems. Sufficient margin must exist to allow for the uncertainties in the rate.

5. The review procedures in the area of control rod patterns and reactivity worths are:

   a. The reviewer determines by inspection and study of the information described in subsection I.5 that the control rod and bank worths are reasonable. This determination involves evaluation of the appropriateness of the analytical models used, the applicability of experimental data used to validate the models, and the applicability of generic positions or those established in previous reviews of similar reactors.

   b. The reviewer determines the equipment, operating restrictions, and administrative procedures that are required to restrict possible control rod and bank worths, and the extent to which the alarm criterion in subsection II.2.c.(2) is satisfied. If the equipment involved is subject to frequent downtime, the reviewer must determine if alternative measures should be provided or the extent of proposed outage time is acceptable.

   c. The reviewer will employ the same procedures as in a, above, to evaluate the scram reactivity information described in subsection I.5. The scram reactivity is a property of the reactor design and is not easily changed, but if restrictions are necessary the procedures in b, above, can be followed as applicable.

6. The information presented on criticality of fuel assemblies is reviewed in the context of the applicant's physics calculations and the ability to calculate criticality of a small number of fuel assemblies. This information is related to information on fuel storage presented in SAR Section 9.1 and reviewed by the Auxiliary Systems Branch (ASB) under SRP Sections 9.1.1 and 9.1.2. The ASB reviewer assumes that the applicant's criticality calculations have been reviewed by CPB and are acceptable.

7. The reviewer exercises professional judgment and experience to ascertain the following about the applicant's analytical methods:

   a. The computer codes used in the nuclear design are described in sufficient detail to enable the reviewer to establish that the theoretical bases, assumptions, and numerical approximations for a given code reflect the current state of the art.

   b. The source of the neutron cross-sections used in fast and thermal spectrum calculations is described in sufficient detail so that the reviewer can confirm

that the cross sections are comparable to those in the current ENDF/B data files (Ref. 6). If modifications and normalization of the cross-section data have been made, the bases used must be determined to be acceptable.

c. The procedures used to generate problem-dependent cross-section sets are given in sufficient detail so that the reviewer can establish that they reflect the state of the art. The reviewer confirms that the methods used for the following calculations are of acceptable accuracy: the fast neutron spectrum calculation; the computation of the U-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example, Pu-240); calculation of the Dancoff correction factor for a given fuel lattice; the thermal neutron spectrum calculation; the lattice cell calculations including fuel rods, control assemblies, lumped burnable poison rods, fuel assemblies, and groups of fuel assemblies; and calculations of fuel and burnable poison depletion and buildup of fission products and transuranium isotopes.

d. The gross spatial flux calculations that are used in the nuclear design are discussed in sufficient detail so that the reviewer can confirm that the following items are adequate to produce results of acceptable accuracy; the method of calculation (e.g., diffusion theory, S_n transport theory, Monte Carlo, synthesis); the number of energy groups used; the number of spatial dimensions (1, 2, or 3) used; the number of spatial mesh intervals, when applicable; and the type of boundary conditions used, when applicable.

e. The calculation of power oscillations and stability indices for diametral xenon reactivity transients, axial xenon reactivity transients, other possible xenon reactivity transients, and non-xenon-induced reactivity transients, are discussed in sufficient detail so that the reviewer can confirm for each item that the method of calculation (e.g., nodal analysis, diffusion theory, transport theory, synthesis) and the number of spatial dimensions used (1, 2, or 3) are acceptable.

f. Verification of the data base, computer codes, and analysis procedures has been made by comparing calculated results with measurements obtained from critical experiments and operating reactors. The reviewer ascertains that the comparisons cover an adequate range for each item and that the conclusions of the applicant are acceptable.

8. The analysis of neutron irradiation of the reactor vessel may be used in two ways. It may provide the design basis for establishing the vessel material nil-ductility transition temperature as a function of the fluence, nvt Or, it may provide the relative flux spectra at various positions between the pressure vessel and the reactor core so that the flux spectra for various test specimens may be estimated. This information is used by the MTEB in determining the reactor vessel material

surveillance program requirements and pressure-temperature limits for operation under SRP Sections 5.3.2 and 5.3.3. CPB reviews the calculational method, the geometric modeling, and the uncertainties in the calculations under this SRP section. The review procedures for pressure vessel irradiation include determinations that:

a. The calculations were performed by higher order theory than diffusion theory.

b. The geometric modeling is detailed enough to properly estimate the relative flux spectra at various positions from the reactor core boundary to the pressure vessel wall.

c. The peak vessel wall fluence for the design life of the plant is less than 10^{20} n/cm^2 for neutrons of energy greater than one MeV. If the peak fluence is found to be greater than this value, the ?EB is notified.

IV. EVALUATION FINDING

The reviewer verifies that sufficient information has been provided and his review supports the following type of evaluation finding, which is to be included in the staff's safety evaluation report:

"The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the _____ plant.

"To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity requirements for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least a ____%Δk/k subcritical margin in the hot condition at any time during the cycle with the highest worth control rod stuck in the fully withdrawn position.

"On the basis of our review, we conclude that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design"; Criterion 11, "Reactor Inherent Protection"; Criterion 12, "Suppression of Reactor Power Oscillations"; Criterion 13, "Instrumentation and Control"; Criterion 20, "Protection System Functions"; Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions"; Criterion 26, "Reactivity Control System Redundancy and Capability"; Criterion 27, "Combined Reactivity Control Systems Capability"; and Criterion 28, "Reactivity Limits."

2. Staff safety evaluation reports and plant technical specifications. Examples of these are:

   a. Safety Evaluation Report, General Electric Standard Safety Analysis Report (GESSAR), Section 4.3, Docket No. STN 50-447, U. S. Atomic Energy Commission, November 1974.

   b. Safety Evaluation Report, Combustion Engineering Standard Safety Analysis Report (CESSAR), Section 4.3, Docket No. STN 50-470, U. S. Nuclear Regulatory Commission, NUREG-75/112, December 1975.

   c. Safety Evaluation Report, Jamesport Nuclear Power Station Units 1 and 2, Section 4.3, Docket Nos. STN 50-516/517, U. S. Nuclear Regulatory Commission, NUREG-75/095, October 1975.

   d. Safety Evaluation Report, Greenwood Energy Center Units 2 and 3, Section 4.3, Docket Nos. 50-452/453, U. S. Atomic Energy Commission, July 17, 1974.

   e. Technical Specifications, Browns Ferry Nuclear Plant Unit 1 and Unit 2, Sections 2.1 and 3.2 through 3.5, License No. DPR-33 and 52, June 28, 1974.

   f. Technical Specifications, Millstone Point Nuclear Power Station Unit No. 2, Sections 2.1, 3.1, and 3.2, Docket No. 50-336, August 1, 1975.

   g. Technical Specifications, D. C. Cook Nuclear Plant Unit 1, Sections 2.1 and 3.1 through 3.3, License No. DPR-58, October 25, 1974.

   h. Technical Specifications, Arkansas Nuclear One Unit 1, Sections 2.1, 3.1, and 3.5, License No. DPR-51, May 21, 1974.

3. Staff technical reports on fuel densification:

   a. Regulatory Staff, "Technical Report on Densification of Light Water Reactor Fuels," U. S. Atomic Energy Commission, November 14, 1972.

   b. Regulatory Staff, "Technical Report on Densification of Babcock and Wilcox Reactor Fuels," U. S. Atomic Energy Commission, July 6, 1973.

c. Regulatory Staff, "Technical Report on Densification of Exxon Nuclear BWR Fuels," U. S. Atomic Energy Commission, September 3, 1973.

d. Regulatory Staff, "Technical Report on Densification of Gulf United Nuclear Fuels Corporation Fuels for Light Water Reactors," U. S. Atomic Energy Commission, November 21, 1973.

e. Regulatory Staff, "Technical Report on Densification of Westinghouse PWR Fuel," U. S. Atomic Energy Commission, May 14, 1974.

4. Topical and startup test reports which are current and applicable to the reactor under review. Examples of these are:

a. G. N. Kear and M. J. Ruderman, "An Analysis of Methods in Control Rod Theory and Comparison with Experiment," GEAP-3937, General Electric Company, May 1962.

b. J. S. Moore, "Power Distribution Control of Westinghouse PWR's," WCAP-7811, Westinghouse Electric Corporation, December 1971.

c. J. O. Cermak, et al., "Pressurized Water Reactors pH-Reactivity," WCAP-3696-8, Westinghouse Electric Corporation, October 1968.

d. "Surry Power Station - Unit 2, Startup Test Report," Virginia Electric Power Company, July 31, 1973.

e. J. E. Outz, "Plant Startup Test Report, H. B. Robinson Unit No. 2," WCAP-7844, Westinghouse Electric Corporation, January 1972.

f. R. H. Clark and T. G. Pitts, "Physics Verification Experiments, Core I," BAW-TM-455, Babcock and Wilcox Company, June 1966.

g. R. H. Clark, "Physics Verification Experiments, Cores II and III," BAW-TM-458, Babcock and Wilcox Company, July 1966.

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A. BACKGROUND

In connection with the staff review of WCAP-8185 (17x17), we reviewed and accepted a scheme developed by Westinghouse for operating reactors that assures that throughout the core cycle including during the most limiting power maneuvers the total peaking factor, F_Q, will not exceed the value consistent with the LOCA or other limiting accident analysis. This operating scheme, called constant axial offset control (CAOC), involves maintaining the axial flux difference within a narrow tolerance band around a burnup-dependent target in an attempt to minimize the variation of the axial distribution of xenon during plant maneuvers.

Originally (early '74), the maximum allowable F_Q (for LOCA) was 2.5 or greater. Later (late '74), when needed changes were made to the ECCS evaluation model, Westinghouse, in order to meet physics analysis commitments to all its customers at virtually the same time, did a generic analysis (one designed to suit a spectrum of operating and soon-to-be-operating reactors) and showed that most plants could meet the requirements of Appendix K and 10 CFR 50.46 (i.e., 2200°F peak clad temperature) if $F_Q \leq 2.32$. Also, Westinghouse showed that CAOC procedures employing a $\pm$ 5% target band would limit peak F_Q for each of these reactors to less than 2.32.

We recognized at that time, however, that not all plants needed to maintain F_Q below 2.32 to meet FAC, or, needed to operate within a $\pm$ 5% band to achieve $F_Q \leq 2.32$. In fact, Point Beach was allowed to operate with a wider band because the Wisconsin Electric Power Company demonstrated to our satisfaction that the reactors could be maneuvered within a wider band (+6,-9%) and still hold F_Q below 2.32. We fully expected that in time most plants would have individual CAOC analyses and procedures tailored to the requirements of their plant-specific ECCS analyses.

Therefore, when we accepted CAOC it was not just $F_Q = 2.32$ and a $\pm$ 5% band width we were approving, but the CAOC methodology. This is analogous to our review and approval of ECCS and fuel performance evaluation models.

The CAOC methodology, which is described in Reference 1, entails (1) establishing an envelope of allowed power shapes and power densities, (2) devising an operating strategy for the cycle which maximizes plant flexibility (maneuvering) and minimizes axial power shape changes, (3) demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics, and (4) demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

Westinghouse argues that point 3, above, is achieved by calculating all of the load-follow maneuvers planned for the proposed cycle and showing that the maximum power densities expected are within limits. These calculations are performed with a radial/axial synthesis method which has been shown to predict conservative power densities when compared to experiment. While we have accepted CAOC on the basis of these analyses, we have also required that power distributions be measured throughout a number of representative (frequently, limiting) maneuvers early in cycle life to confirm that peaking factors are no greater than predicted. Additionally, we are sponsoring a series of calculations at BNL to check aspects of the Westinghouse analysis.

The power distribution measurement tests described above will, of course, automatically relate incore and excore detector responses, and thereby validate that power distribution control can be managed with excore detectors.

B. BRANCH TECHNICAL POSITION

An applicant or licensee proposing CAOC for other than $F_Q = 2.32$ and $\Delta I = \pm 5\%$ is expected to provide:

1. Analyses of F_Q x power fraction showing the maximum $F_Q(z)$ at power levels up to 100% and DNB performance with allowed axial shapes relative to the design bases for overpower and loss of flow transients. The envelope of these analyses must be shown to be valid for all normal operating modes and anticipated reactor conditions. (See Table 1 of Reference 2 for the cases which must be analyzed to form such an envelope.)

2. A description of the codes used, how cross-sections for cycle were determined, and what F_{xy} values were used.

3. A commitment to perform load-follow tests wherein F_Q is determined by taking incore maps during the transient. (NOTE: Westinghouse has outlined for both the NRC staff and the ACRS an augmented startup test program designed to confirm experimentally the predicted power shapes. This program is presented in a Westinghouse report (Ref. 3). The tests will be carried out at several representative - both 15x15 and 17x17 - reactors. We have endorsed these tests as has the ACRS in its June 12, 1975 letter for the Diablo Canyon plant. In addition, for the near term, we plan to require that those licensees who propose to depart from the previously approved peaking factor and target band width perform similar tests, precisely which ones to be determined on a case-by-case basis, to broaden our confidence in analytical methods by extending the comparison of prediction with measurement to include more and more burnup histories.)

C. REFERENCES

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SECTION 4.4 . THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The objectives of the review are to confirm that the thermal and hydraulic design of the core and the reactor coolant system (RCS) has been accomplished using acceptable analytical methods; is equivalent to or is a justified extrapolation from proven designs; provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational transients; and is not susceptible to thermal-hydraulic instability. This plan describes the normal review of thermal and hydraulic design, i.e., that for a plant similar in core and primary coolant system design to previously reviewed plants. The review of new prototype plants or new design methods require in addition the independent audit analyses discussed in the appendix to this plan.

The review includes evaluation of the proposed technical specifications regarding safety limits and limiting safety system settings, to ascertain that these are consistent with the power-flow operating map for boiling water reactor (BWR) plants or the temperature-power operating map for pressurized water reactor (PWR) plants.

The review also includes determination of the largest hydraulic loads on core and reactor coolant system components during normal operation and postulated accident conditions. This information is provided to the Mechanical Engineering Branch for use in the review of reactor components and structures.

To accomplish the objectives, the reviewer examines features of core and RCS components, key process variables for the coolant system, calculated parameters characterizing thermal performance, data serving to support new correlations or changes in accepted correlations, and assumptions in the equations and solution techniques used in the analyses. The reviewer determines that the applicant has used approved analysis methods in the manner specified by topical reports describing the methods and by staff reports approving the methods. If an applicant has used previously unapproved correlations or analysis methods, the reviewer initiates a generic evaluation.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

The CPB, as described in Standard Review Plan (SRP) 4.3, provides technical consultation on matters related to core physics calculations and their integration with power distribution assumptions made for the core thermal and hydraulic analysis. The CPB, on request, participates in generic evaluation of new thermal and hydraulic analysis methods.

II. ACCEPTANCE CRITERIA

1. General Design Criterion 10 (Ref. 1) requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   There are two acceptable approaches to meet this criterion:

   a. For departure from nucleate boiling ratio (DNBR) or critical heat flux ratio (CHFR) correlations there should be a 95% probability, at the 95% confidence level, that no fuel rod in the core experiences a departure from nucleate boiling condition during normal operation or transients that are anticipated to occur with moderate frequency.

   b. For critical power ratio (CPR) correlations, the limiting (minimum) value of CPR is to be established such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition during core wide transients. For transients that effect only a portion of the core, the same value of CPR is used to provide additional conservatism.

   Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions. As guidance to the reviewer, the correlations listed below have been found acceptable for previously reviewed plants.

   a. BWR's - The minimum CHFR calculated with the Hench - Levy correlation (Ref. 8) should exceed 1.0 at all times. The value of the minimum CPR calculated with the GETAB analysis will vary for different plants and/or product lines. Typically, the value will exceed 1.06.

   b. PWR's - For 14 x 14 or 15 x 15 rod arrays the minimum DNBR calculated with due allowance for mixing grids (Refs. 3, 4, and 5) should exceed 1.32 using the BAW-2 correlation (Ref. 6) and 1.30 using the W-3 correlation (Ref. 7).

2. As problems affecting DNBR or CPR limits arise, such as fuel densification or rod bowing, experimental and/or analytical methods for determining the appropriate design penalties are included in the review. Subchannel hydraulic analysis codes such as those described in References 8, 9, and 10 should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments

or power reactor cores. Calculations of BWR fluid conditions for use in CHF correlations have been in accordance with the models specified in Reference 11.

3. The maximum value of the linear heat generation rate (LHGR) anywhere in the core, including all hot spot and hot channel factors, should be such that the centerline temperature of the fuel is below the melting temperature. For most core designs, full power steady-state operation is not the operating mode which is most limiting in regard to LHGR. Rather, ECCS performance following a postulated loss-of-coolant accident or various anticipated transients is more limiting. As guidance for the reviewer, the following values of LHGR have been found acceptable for previously reviewed designs:

LHGR (kW/ft)

BWR	PWR
17.5 kW/ft for 1965 product line	18.5 kW/ft for 15 x 15 array
18.5 kW/ft for 1967 product line	18.5 kW/ft for 14 x 14 array
18.5 kW/ft for 1969 product line	13.0 kW/ft for 17 x 17 array
13.4 kW/ft for BWR-6	13.0 kW/ft for 16 x 16 array

While these values do not constitute criteria for acceptance, any design in which they are exceeded must be supported by sufficient analysis to demonstrate that all acceptance criteria are met. Other operating conditions such as fuel densification may reduce these values.

4. For PWR and BWR fuel, the maximum clad strain calculated for operational transients and at end-of-life should be less than one percent. These analyses should consider the pressure associated with gaseous fission products.

5. The reactor should be demonstrated to be free of undamped oscillations or other hydraulic instabilities for all conditions of steady-state operation, for all operational transients, for all load-following maneuvers, and for partial loop operation. Typical methodologies are described in References 12 and 13.

6. Methods for calculating single-phase and two-phase fluid flow in the RCS piping and other components should include classical fluid mechanics relationships and appropriate empirical correlations. For components of unusual geometry, such as the following, these relationships should be confirmed empirically:

   a. Steam generator (Ref. 14).

   b. Reactor vessel (Ref. 15).

   c. Jet pump (Ref. 16).

   d. Core flow distribution (Refs. 15, 17, and 18).

7. The proposed technical specifications should be established such that the plant can be safely operated at steady state conditions under all of the expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, such that acceptance criterion 1, above, is satisfied. The limiting conditions of operation must provide appropriate operating restrictions. For example, the limiting conditions of operation must assure that the reactor coolant pumps have adequate net positive suction head for all expected modes of operation.

8. Any changes to accepted codes, correlations, and analytical procedures, or the addition of new ones must be justified on theoretical or empirical grounds.

9. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68 (Ref. 19), as regards measurements, and confirmation of thermal hydraulic design aspects.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this review plan. For operating license (OL) applications, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to safety limits, limiting safety system settings, and conditions of operation.

The reviewer must begin with an understanding of currently acceptable thermal and hydraulic design practice for the reactor type under review. This understanding can be most readily gained from topical reports describing CHF correlations, system hydraulic models and tests, and core subchannel analysis methods; from standard texts and other technical literature which establish the methodology and the nomenclature of this technology; and from documents which summarize current staff positions concerning acceptable design methods.

Much of the review described below is generic in nature and is not performed for each plant. That is, the RSB reviewer is to compare the core design and operating parameters to those of previously reviewed plants. He then devotes the major portion of his review effort to those areas where the application is not identical to previously reviewed plants.

The reviewer is to compare the information in the applicant's safety analysis report (SAR) to the documents referenced by the applicant or in this plan to determine conformance to the bounds established by such documents. The reviewer must confirm that void, pressure drop, and heat transfer correlations used to estimate fluid conditions (flow, pressure, quality) are within the ranges of applicability specified by their authors or in previous staff reviews, that the analysis methods are used in the manner specified by the developers or in previous staff reviews, that the reactor design falls within the ranges of applicability

specified for accepted analysis methods, and that the design is within the criteria specified in II, above, and is not an unexplained or unwarranted extrapolation of other thermal-hydraulic designs.

The review does not routinely involve calculations by the staff. On occasion, e.g., if a new model or correlation is proposed, independent analyses are performed by the staff or by consultants under the direction of the RSB. These analyses establish the range of applicability and associated accuracy of the new model or correlation and the reviewer ensures it is applied accordingly.

The reviewer is to establish that the thermal-hydraulic design and its characterization by MCHFR or DNBR have been accomplished and are presented in a manner which accounts for all possible reactor operating states as determined from operating maps. In this regard, the reviewer must confirm, with the aid of the CPB, that the power distribution assumptions of SAR Section 4.4 are a conservative (i.e. worst-case) accounting of the power distributions derived in Section 4.3 from core physics analyses, and that the latter analyses include an acceptable calculation of local void fractions. He must also confirm that the mass flux used in these calculations takes into account the core flow distribution (including that for partial loop operation) and the worst case of core bypass flow. The reviewer confirms that the primary coolant flow range shown in the operating map will be verified by pre-startup measurements.

The reviewer ensures that adequate account is taken of the effect of crud in the primary coolant system, such as in the calculation of CHF in the core, heat transfer in the steam generators, and pressure drop throughout the RCS.

The reviewer is to examine the calculation of hydraulic loads for normal operations and postulated accidents, ensure they are accurately estimated for the worst cases, and supply the worst case values to the Mechanical Engineering Branch for their review of reactor components and supports.

The reviewer should be aware of the vibration and loose-parts monitoring equipment and procedures used on other comparable plants and, taking into account pertinent differences, ensure that an adequate system is provided for the plant under review.

The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 19). At the OL stage, the reviewer is to assure that sufficient information is provided by the applicant to identify clearly the test objectives, methods of testing, and acceptance criteria. (See par. C.2.b of Reference 19.)

The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the core and reactor coolant system will satisfy functional requirements. As an alternative to this detailed evaluation, the reviewer may compare the core and reactor coolant system design to that of previously reviewed plants. If the design is essentially

identical and if the proposed test programs are essentially the same as performed previously on other plants, the reviewer may conclude that the proposed test programs are adequate for the core and reactor coolant system.

If the core or the reactor coolant system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the preoperational and initial startup testing programs are reviewed at the construction permit stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

The proposed technical specifications that relate to the core and the reactor coolant system are evaluated. This evaluation is to cover all of the safety limits and bases that could affect the thermal and hydraulic performance of the core. The limiting safety system settings are reviewed to ascertain that acceptable margins exist between the values at which reactor trip occurs automatically for each parameter (or combinations of parameters) and the safety limits. The reviewer confirms that the limiting safety system settings and limiting conditions for operation, as they relate to the reactor coolant system, do not permit operation with any expected combination of parameters that would not satisfy criterion 1 of Section II.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

> "The thermal-hydraulic design of the core for the _____ plant was reviewed. The scope of review included the design criteria, preliminary core design, and the steady-state analysis of the core thermal-hydraulic performance.* The review concentrated on the differences between the proposed core design (and criteria) and those designs and criteria that have been previously reviewed and found acceptable by the staff. It was found that all such differences were satisfactorily justified by the applicant. The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.
>
> "The staff concludes that the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable regulatory guides and staff technical positions and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

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9. H. Chelemer, P. T. Chu, and L. E. Hochreiter, "THINC-IV-An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, Westinghouse Electric Corporation, June 1973 (under review). (See also WCAP-7359-L and WCAP-7838.)

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11. B. C. Slifer and J. E. Hench, "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, Appendix C, General Electric Company, April 1971.

12. L. A. Carmichael and G. J. Scatena, "Stability and Dynamic Performance of the General Electric Boiling Water Reactors," APED-5652, General Electric Company, April 1969.

13. R. L. Rosenthal, "An Experimental Investigation of the Effect of Open Channel Flow on Thermal-Hydrodynamic Flow Instability," WCAP-7966, Westinghouse Electric Corporation, October 1968.

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15. B. S. Mullanax, R. J. Walker and B. A. Karrasch, "Reactor Vessel Model Flow Tests," BAW-10037 (non-proprietary version of BAW-10012), Revision 2, Babcock and Wilcox Company, September 1968.

16. "Design and Performance of General Electric Boiling Water Reactor Jet Pumps," APED-5460, General Electric Company, September 1968.

17. H. T. Kim, "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299, General Electric Company, January 1971.

18. F. D. Carter, "Inlet Orificing of Oper PWR Cores," WCAP-9004, Westinghouse Electric Corporation, January 1969.

19. Regulatory Guide No. 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

APPENDIX

STANDARD REVIEW PLAN 4.4

INDEPENDENT AUDIT ANALYSIS

The Core Performance Branch may be requested to perform independent analyses of thermal-hydraulic and physics phenomena for both steady-state and transient conditions. These analyses may be requested by various technical groups within the staff.

The required analyses may be in the following forms:

1. Independent computer calculations to substantiate reactor vendor analyses of steady-state or transient events.

2. Evaluations of vendor computer programs and analysis methods.

3. Reductions and correlations of experimental data to verify processes or phenomena which are applied to reactor design. These independent audit analyses may also be undertaken in support of Standard Review Plans (SRP) 4.2 and 4.3, in addition to the independent analysis discussed in SRP 4.4.

TYPES OF ANALYSES

The types of analyses that are performed are the following:

1. Steady-State Analyses

   a. The steady-state reactor core flow distribution, steam quality, void distribution, and pressure drop have been calculated for PWR-type fuel assemblies using the multichannel boiling code, COBRA III-C (Ref. 1). From these quantities COBRA III-C also calculates the fuel thermal margin in terms of the ratio of the local predicted critical heat flux to the operating heat flux. The W-3 and B&W-2 critical heat flux correlations (Refs. 2 and 3) are used in the code. From these results, the thermal margin and fuel clad temperature calculated by the vendor's computer program can be verified. To the extent possible, inputs to computer programs used by the staff correspond to those used by the reactor vendors.

   b. Through the use of consultants, independent comparisons and correlations are made of data from experimental programs. These reviews also include analyses of experimental techniques, test repeatability, and data reduction methods.

2. Transient Analyses

   Independent computer calculations are performed to provide an audit on the adequacy of a particular analysis performed by an applicant. The thermal-hydraulic phenomena associated with the transient are calculated with the RELAP-3 (Ref. 4) or RELAP-4 (Ref. 5) computer programs. The fuel performance is calculated by the COBRA III-C program, which obtains the necessary thermal-hydraulic parameters from the above programs.

3. Loss-of-Coolant Accident (LOCA) Analyses

Independent calculations are performed by the staff to verify the LOCA analyses submitted by applicants in accordance with the requirements of 10 CFR §50.46 and Appendix K of 10 CFR Part 50. These calculations are performed to check the blowdown phenomena, ECCS response, and fuel cladding temperature transients. The ECCS performance criteria are specified in Appendix K. Also, sensitivity studies are performed to verify the convergence of analytical techniques, and the sensitivity to various postulated break sizes, types, and locations.

Evaluations are also made of the computer programs used by the vendors to perform ECCS evaluations. These computer programs are checked to determine conformance with the required features specified in Appendix K. In addition, the analysis methods and heat transfer correlations are evaluated by comparison with existing experimental data.

4. Reactivity Analyses

Independent analyses are performed by consultants to provide a check on the adequacy of a particular analytical method and the basic assumptions. These include items such as maximum control rod worth, power distribution, and reactivity coefficients such as the Doppler and moderator temperature coefficients.

Staff consultants assess the conservatism of the vendors' models, either by comparison with experiment or with more sophisticated models. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes are investigated and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant are investigated.

5. ATWS (Anticipated Transients Without Scram) Analyses

Independent audit analysis for ATWS serves three specific functions:

a. To confirm the vendors' and applicants' interpretations of WASH-1270 (Ref. 6).

b. To evaluate the adequacy of backup protection systems.

c. To confirm the accuracy of the calculation of consequences of ATWS and of the models used in the analysis.

The RELAP 3B computer code (Ref. 7) will be used by the staff and its consultants for the ATWS studies. The preparation of data for an independent audit computation requires a careful review of all reactor systems to ascertain if operational credit can be taken for them in the analysis. The process will then serve as a means of confirming the vendors' and applicants' interpretations of WASH-1270.

An evaluation of the effectiveness and the response of a backup protection system is achieved by an audit computation. The degree of protection can be evaluated by conducting analyses with and without the backup system.

A calculation of the consequences of an ATWS transient serves as a means of evaluating vendors' analytical models and the accuracy of the results. For example, an ATWS loss of load event for a PWR can result in very high primary pressure. The magnitude of the pressure response is a function of the performance of the pressurizer safety valves in discharging water. An independent audit computation would verify the analytical model used for discharging water and the magnitude of the pressure response. The pressure response is important in evaluating the integrity of the reactor vessel.

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SECTION 4.5.1 CONTROL ROD DRIVE STRUCTURAL MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

General Design Criterion 26 requires that one of the reactivity control systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences. The areas listed below relating to materials considerations in the design of the control rod drive mechanism are reviewed. The review areas are similar to those given in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials." For the purpose of this plan, the control rod system is comprised of the control rod drive mechanism (CRDM) and extends only to the coupling interface with the reactivity control (poison) elements in the reactor vessel; it does not include the electrical and hydraulic systems necessary for actuating the CRDMs.

The mechanical aspects of the control rod system other than the reactivity control elements are reviewed by the Mechanical Engineering Branch in accordance with Standard Review Plan 3.9.4.

The mechanical design, thermal performance and chemical compatibility of the reactivity control elements are addressed by the Core Performance Branch in accordance with Standard Review Plan 4.2.

1. Materials Specifications

The properties of the materials used in the control rod drive are reviewed from the standpoint of adequate performance throughout the design life of the plant (or component). Materials commonly used include austentic stainless steels (which may be cold worked) chromium-plated stainless steels, martensitic stainless steels, precipitation-hardening stainless steels such as 17-4 PH, and other special-purpose materials such as cobalt-base alloys (Stellites), Inconel-750, Colmonoy-6 and Graphitar-14.

2. Austentic Stainless Steel Components

Areas of review for austenitic stainless steel components are similar to SRP 5.2.3 Sec. I.4.a,b,d and e.

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The use of sensitized stainless steels should be controlled to prevent stress-corrosion cracking of the material during operation of the plant. Welding procedures should be controlled to reduce the probability of sensitization and microfissure formation. Cold-worked stainless steels should not have high yield stress, to reduce the probability of stress-corrosion cracking during operation of the plant.

3. Other Materials

Special requirements for the other materials include tempering and aging temperatures for martensitic and precipitation-hardening stainless steels to prevent their deterioration by stress corrosion during operation of the plant. The compatibility of these materials with the reactor coolant is reviewed to assure that they will continue to perform satisfactorily throughout the life of the component.

Metallic and non-metallic materials used in the control rod drive mechanism and not included in Appendix I to Section III of the ASME B&PV Code are identified.

4. Cleaning and Cleanliness Control

Proper care should be taken in handling the materials and parts of the control rod drive mechanism during fabrication, shipping, and onsite storage to assure that all cleaning solutions, processing compounds, degreasing agents, and other foreign materials are completely removed, and that all parts are dried and properly protected following any flushing treatment with water.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review listed in Section I of this plan are as follows:

1. Materials Specifications

The properties of the materials selected for the control rod drive mechanism must be equivalent to those given in Appendix I to Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), or Parts A and B of Section II of the Code, except that cold-worked austenitic stainless steels shall have a 0.2% offset yield strength no greater than 90,000 psi, to reduce the probability of stress corrosion cracking occurring in these systems. Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials," describes the acceptable code cases that may be used in conjunction with the above specifications.

2. Austentic Stainless Steel Components

Acceptance criteria used are similar to SRP 5.2.3 Sec. II.4.a,b,d and e.

Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," describes acceptance methods for preventing integranular corrosion of stainless steel

components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," describes acceptance criteria for assuring the integrity of welds in stainless steel components of these systems.

3. Other Materials

All materials for use in this system must be selected for their compatibility with the reactor coolant, as described in Articles NB-2160 and NB-3120 of the Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardening stainless steels should be specified to provide assurance that these materials will not deteriorate because of stress corrosion cracking in service. Acceptable heat treatment temperatures include aging at 1050° - 1100°F for Type 17-4 PH and 1050°F for Type 410 stainless.

4. Cleaning and Cleanliness Control

Onsite cleaning and cleanliness control should be in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI N 45.2 1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants." The oxygen content of the water in vented tanks is not required to be controlled. Vented tanks with deionized or demineralized water are a normal source of water for final cleaning or flushing of finished surfaces. Halogenated hydrocarbon cleaning agents should not be used.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case. To ascertain that the acceptance criteria given in Section II are met, the reviewer examines the review areas listed in Section I for the required information, using the following procedures:

1. Materials Specifications

The reviewer compares the properties of the materials proposed for the control rod system with Appendix I to Section III of the Code, and Parts A, B and C of Section II of the Code. He verifies that cold-worked austenitic stainless steels used in fabrication of the reactivity control mechanisms are in conformance with Section II.1, above.

2. Austenitic Stainless Steel Components

Review procedures are similar to those in SRP 5.2.3 Sec. III.4.a, b, d and e. The methods of controlling sensitized stainless steel are examined by the reviewer and compared with the positions given in Regulatory Guide 1.44, especially with

respect to cleaning and protection from contamination during handling and storage, verification of non-sensitization of the material, and qualification of welding processes employed in production using ASTM A-262. If alternative methods of testing the qualification welds for degree of sensitization are proposed by the applicant, the reviewer determines if these are satisfactory, taking into account branch positions taken on previous applications and the degree of equivalence of the alternate methods. The reviewer may ask the applicant to justify the technical basis for any departures from the cited positions. Alternative tests that have been accepted by the branch include the use of ASTM A-708.

The methods of controlling and measuring the amount of delta ferrite in stainless steel weld deposits are examined by the reviewer and compared to the positions in Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," especially with respect to the filler metal acceptance procedures for the determination of delta ferrite content. If alternative positions are proposed by the applicant, the reviewer determines if these are satisfactory, taking into account branch positions taken on previous applications. The reviewer may ask the applicant to justify the technical basis for any departures from the acceptance criteria stated in Section II.2.

3. Other Materials

The reviewer examines the information provided in the applicant's safety analysis report (SAR) on the compatibility of the materials (other than austenitic stainless steels) to be used in contact with the reactor coolant. He determines that the materials are compatible with the service environment so that corrosion or stress corrosion of the component will not occur during the lifetime of the component. Metallic and nonmetallic materials identified in Section I.3 are reviewed to assure compatibility and that loss of integrity will not occur during the life of the component.

The reviewer determines that the tempering temperatures of all martensitic stainless steels and the aging temperatures of precipitation-hardening stainless steels have been specified, and are in accordance with the acceptance criteria in Section II.3.

4. Cleaning and Cleanliness Control

The reviewer verifies that onsite cleaning and cleanliness control procedures are satisfactory and in accordance with Section II.4.

5. General

If the information contained in the SAR or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional

information, but also, the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the acceptance criteria.

IV. EVALUATION FINDINGS

When the reviewer has verified that sufficient and acceptable information has been provided in accordance with the requirements of this review plan, conclusions of the following type are prepared, to be included in the staff's safety evaluation report:

"The properties of materials selected for the control rod drive mechanism components exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code, and Part A, B and C of Section II of the Code, and conform with the staff position that the yield strength of cold-worked austenitic stainless steel should not exceed 90,000 psi."

"The controls imposed upon the austenitic stainless steel of the mechanisms conform to the recommendations of Regulatory Guide 1.31, "Controls of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the component. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with staff positions. Cleaning and cleanliness control are in accordance with ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plant."

"Conformance with the codes, standards, and Regulatory Guides indicated above, and conformance with the staff positions on the allowable maximum yield strength of cold-worked austenitic stainless steel, and the tempering or aging temperatures of martensitic and precipitation-hardened stainless steels constitute an acceptable basis for meeting in part the requirements of General Design Criterion 26."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

2. ASME Boiler and Pressure Vessel Code, Section III, Appendix I, and Section II, Parts A, B and C, American Society of Mechanical Engineers.

3. ASTM A-262, "Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel Practice A and E," Annual Book of ASTM Standards, American Society for Testing and Materials.

4. ASTM A-708, "Detection of Susceptibility of Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel."

5. ANSI N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants, "American National Standards Institute.

6. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

7. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

8. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

9. Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials."

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SECTION 4.5.2 REACTOR INTERNALS MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 1 and 14 require that structures, systems, and components important to safety shall be designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The following areas in the applicant's safety analysis report (SAR) relating to reactor internals materials are reviewed:

1. Material Specifications

The review includes the specifications for the materials, including weld materials, to be used for major components of the reactor internals which consist of the core support structures and the internal structures. For boiling water reactors (BWR's), these specifications should include materials for shrouds, shroud supports, top guides, fuel support pieces, control rod drive tubes, jet pump assemblies, shroud head and steam separator assembly, and steam dryers. For pressurized water reactors (PWR's), the specifications should include materials for the lower core support structures, including the core barrel, neutron shield pad assembly, core baffle, lower core plate, and core supports, and materials for the upper core support structures including the top support plate, beam sections, upper core plate, support columns, and guide tube assemblies, and the in-core instrumentation support structure.

The adequacy and suitability of the materials specified for the above applications are reviewed in terms of their mechanical properties, stress corrosion resistance, and fabricability.

2. Controls on Welding

The review includes the controls on welding of materials used for reactor internals.

3. Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

The review includes the information submitted by the applicant on the nondestructive examinations used for inspection of the subject product forms.

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4. Austenitic Stainless Steel

Austenitic stainless steels, in a variety of product forms, are used for construction of components in the reactor internals. Unstabilized austenitic type stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316, are normally used.

Since these compositions are susceptible to stress corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid severe sensitization of the material, and to minimize exposure of the stainless steel to contaminants that could lead to stress-corrosion cracking. The review includes information submitted by the applicant in these areas, as described in SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

5. Other Materials

Materials other than austenitic stainless steels are reviewed in terms of their properties, stress corrosion resistance and fabricability.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Subsection I of this plan are as follows:

1. Material Specifications

For core support structures and reactor internal structures, the permitted material specifications are those given in the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, NG-2121, and Tables 1-1.1 and 1-1.2 of Appendix I. These materials are described in detail in the Code, Section II, Parts A, B, and C.

Additional permitted materials are those shown in ASME Code Cases approved for use by Regulatory Guide 1.85, "Code Case Acceptability, ASME Section III Materials."

2. Controls on Welding

The welds of components for core support structures and reactor internals are fabricated in accordance with the Code, Section III, NG-4400, and must meet the examination and acceptance criteria shown in NG-5000.

3. Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

Examination shall be in accordance with the requirements of ASME Code Section III, NG-2550.

The acceptance criteria shall be in accordance with the requirements of ASME Code Section III, NG-5300.

4. Austenitic Stainless Steels

The acceptance criteria for this area of review are given in Section II.4 of SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

5. Other Materials

All materials used for reactor internals must be selected for their compatibility with the reactor coolant, as described in Subarticles NG-2160 and NG-3120 of the Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are aging at 1050°F - 1100°F for Type 17-4 PH and tempering at 1050°F for Type 410 stainless steel.

Other materials shall have similar appropriate heat treat and fabrication controls in accordance with strength and compatibility requirements.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For the areas of review described in Subsection I of this plan, the review procedure is as follows:

1. Material Specifications

The list of the materials of construction of the components of the reactor internals that are exposed to the reactor coolant is reviewed.

The material specifications for each component or part used in the reactor internals are compared with the acceptable specifications listed in the Code, Sections II and III, as shown in the acceptance criteria. Any exceptions to the Code materials specifications are clearly identified. The reviewer evaluates the basis for the exceptions, taking into account precedents set in earlier cases, and determines the acceptability of such materials.

2. Controls on Welding

The information submitted by the applicant is reviewed to provide assurance that welding of materials used for components of the reactor internals is in accordance with the procedures of the Code, Section III, NG-4400. The controls on welding, discussed in SRP Section 5.2.3, are considered applicable to welding of reactor internals, and information in this area is verified. The reviewer assures that any special welding process or welding control conforms to the qualification requirements of the Code, Section IX, or that justification is made for any deviation.

3. Nondestructive Examination of Wrought Seamless Tubular Products

The information submitted by the applicant is reviewed to determine methods used for nondestructive examination. The Code, Section III, NG-2550(d) specifies that examination by either radiographic or ultrasonic examination is acceptable.

4. Austenitic Stainless Steel

The materials and fabrication procedures used for reactor internals are reviewed. The areas of review and review procedures follow closely those spelled out in SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials." Environmental conditions must be controlled and welding procedures must be such that the probability of sensitization and microfissuring is reduced. In addition, the reviewer verifies that the material and reactor coolant compositions have been selected to assure compatibility, and that the fabrication and cleaning controls imposed on stainless steel components are designed to prevent contamination with chloride and fluoride ions.

5. Other Materials

The material specifications and fabrication procedures are reviewed to verify that the heat treatment and welding controls are appropriate for the material. The reviewer verifies that the fabrication and cleaning controls will preclude contamination of nickel base alloys by chloride ions, fluoride ions or lead.

6. Additional Information Request

If the information contained in the SAR does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, the reviewer prepares a request for additional information for transmittal to Project Management. Such requests not only identify the additional information required, but also specify the changes needed in the SAR or the plant Technical Specifications to meet acceptance criteria. Subsequent amendments received in response to these requests are reviewed for compliance with the acceptance criteria.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the SRP section, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used for construction of components of the reactor internals have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code.

"The materials for reactor internals exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion on all materials is expected to be negligible.

"The controls imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

"The controls imposed upon components constructed of austenitic stainless steel, as used in the reactor internals, satisfy the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal, and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the materials used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with these regulatory guides constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 14."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants."

2. ASME Boiler and Pressure Vessel Code, Section II, Parts A, B, and C, Section III, and Section IX, American Society of Mechanical Engineers.

3. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

4. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

5. SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

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SECTION 4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Mechanical Engineering Branch (MEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The RSB reviews the combined functional performance of all the reactivity control systems to confirm that the systems can effect a safe shutdown, respond within acceptable limits during anticipated transients, and prevent or mitigate the consequences of postulated accidents.

The reactivity control systems whose functional performance is reviewed by the RSB include: control rod drive system (CRDS), chemical and volume control system (CVCS) for pressurized water reactors (PWR's), standby liquid control system (SLCS) for boiling water reactors (BWR's) and the recirculation flow control system (RFCS) fo. ?WR's. Other aspects of each of these systems are evaluated by other reviewers as noted below.

The CPB in Standard Review Plan (SRP) 4.3 verifies the reactivity control requirements of the combined reactivity control systems. The negative reactivity available in the reactivity control systems, the allowable reactivity insertion or withdrawal rates, and the values of reactivity parameters throughout plant life are evaluated. Matters relating to steady-state core physics calculations and their integration with power distribution assumptions are considered in the CPB review.

The EICSB reviews in SRP 7.7 the control system for the RFCS. The intent of the EICSB review is to assure that failures of the control system would not impair the protection system capability in any significant manner. The EICSB also assists the RSB in reviewing the time delays for the actuation of each of the reactivity control systems. The EICSB in SRP 7.2 evaluates the results of failure modes and effects analyses to assure that a single failure occurring in the control system, or an operator error, will not result in the loss of capability for safe shutdown.

The APCSB, with the aid of the CPB reviewer, reviews the functional capability of the CVCS (for PWR's) and the SLCS (for BWR's) in SRP 9.3.4 and SRP 9.3.5, respectively, to determine

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the adequacy of each system to perform its function of reactivity control for the reactor.

The MEB reviews in SRP 3.9.4 the CRDS to evaluate the adequacy of the system to perform its mechanical function (e.g., rod insertion and withdrawal, scram operation and time) and to maintain the reactor coolant pressure boundary. The pressure-containing components of the CRDS are reviewed by the RSB in SRP 3.2.1 and SRP 3.2.2 to determine that design code requirements, as applicable to the assigned safety class and seismic category, are met.

II. ACCEPTANCE CRITERIA

Acceptability of the information presented in Section 4.6 of the applicant's safety analysis report (SAR), including related sections, is based on meeting the general design criteria (Ref. 1). The acceptance criteria for the areas of review are the following:

1. General Design Criterion 20, "Protection System Functions," as related to the automatic actuation of the reactivity control systems in accident conditions.

2. General Design Criterion 21, "Protection System Reliability and Testability," as related to system design requirements for high functional reliability and capability to meet the single failure criterion.

3. General Design Criterion 23, "Protection System Failure Modes," as related to failing into a safe state.

4. General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions," as related to the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded for malfunction of any reactivity control system.

5. General Design Criterion 26, "Reactivity Control System Redundancy and Capability," as related to the capability of the reactivity control system to regulate the rate of reactivity changes resulting from operational occurrences.

6. General Design Criterion 27, "Combined Reactivity Control Systems Capability," as related to the combined capability of reactivity control systems and emergency core cooling systems to cool the core under accident conditions.

7. General Design Criterion 28, "Reactivity Limits," as related to postulated reactivity accidents.

III. REVIEW PROCECURES

The RSB reviewer evaluates the capabilities of the combined operation of the reactivity control systems to effect reactor shutdown for all postulated operating conditions.

The review procedures set forth below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set

forth in the applicant's preliminary safety analysis report (PSAR) meet the acceptance criteria given in Section II of this review plan. During the operating license (OL) review, the reviewer verifies that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report (FSAR).

1. The RSB reviews the CRDS design with respect to fluid systems and possible single failures. The review of the system description includes piping and instrumentation diagrams (P&IDs), layout drawings, process flow diagrams, and descriptive information on essential supporting systems. The SAR is reviewed to ascertain that failure modes and effects analyses have been completed to determine that the control rod drive system (not the individual drives) is capable of performing its safety-related function following the loss of any active component. The RSB reviewer further confirms, on the basis of previously approved systems or independent failure modes and effects analyses, that the minimum system requirements are met for the failure conditions.

2. The CRDS, P&IDs, layout drawings, and component description and characterictics are reviewed by the RSB to verify that essential portions of the system are correctly identified and are isolable from non-essential portions. The essential portions should be protected from the effects of high or moderate energy line breaks. Layout drawings of the system are reviewed to assure that no high or moderate energy piping systems are close to the CRDS, or that protection is provided from the effects of high or moderate energy pipe breaks.

3. For plants containing control rod drive cooling systems (e.g., using air or water as coolant), the description and drawings are reviewed to determine that the systems meet the design requirements. Essential equipment should be delineated in the SAR. The major function of the cooling system in PWR's is to cool the drive mechanism and remove heat from the CRDS motors to preclude motor burnout or damage. Failure of a CRDS motor could result in a rod drop. In BWR's, the major function of the cooling water is to cool the drive mechanism and its seals to preclude damage resulting from long-term exposure to reactor temperatures. The control rod drive hydraulic system includes the cooling function as part of its design. The RSB reviewer confirms by failure modes and effects analysis that the cooling system is capable of maintaining the CRDS temperature below the applicant's maximum temperature criterion. The EICSB reviewer in SRP 7.2 confirms that there are sufficient instrumentation and controls available to the reactor operator to provide information in the control room to monitor the CRDS conditions, including the more significant parameters such as coolant flow, temperature, and pressure and stator temperature.

4. In coordination with the MEB, the RSB reviews the functional tests of the CRDS as related to rod insertion and withdrawal and scram operation and time. The reviewers check the elements of the test program to ensure that all required thermal-hydraulic conditions have been included for all postulated operating conditions. Experimental

verification of system operation where a single failure has been assumed should be included in the test program, e.g., accumulator leakage for hydraulic CRDS and stuck rod operation.

5. The applicant's proposed preoperational and initial startup test programs are reviewed to determine if they provide reasonable assurance that the CRDS will perform its safety function. This aspect of the review is to verify that sufficient information is provided to identify the test objectives, methods of testing, and test acceptance criteria. If the design is essentially identical and if the proposed test programs are essentially the same as those of previously reviewed plants, the reviewer may conclude that the proposed test programs are adequate. If the proposed CRDS differs from that of prior designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are evaluated.

6. The plant technical specifications are reviewed by the RSB as follows:

   a. For CP's, the reviewer confirms the suitability of the limiting conditions of operation to ensure that the specified operating parameters (scram time, CRDS temperature, operation with inoperable rods) are within the bounds of the analyzed conditions.

   b. For OL's, the reviewer confirms that the content and intent of the technical specifications proposed by the applicant are in agreement with the requirements developed as a result of the staff's review. Where necessary, the review will include requirements for system functional testing, minimum performance, and surveillance requirements.

   c. The reviewer verifies by comparison with other plant reviews that the frequency and scope of periodic surveillance testing is adequate.

7. The reactivity control systems are evaluated to verify that redundant reactivity control systems are not vulnerable to common mode failures. The RSB identifies the common mode failures and the EICSB, MEB, and APCSB assist the RSB reviewer in connection with their responsibilities in SRP 7.4, 3.9.4, and 9.3.4 or 9.3.5, respectively. In addition, the reviewer determines that inadvertent operation of any component or system (e.g., inadvertent scram of axial power shaping rods or inadvertent dilution of boron concentration) would not cause degraded system conditions beyond the capabilities of the safety systems.

8. The RSB reviewer examines all transients and accidents in Chapter 15 of the SAR that require reactivity control systems to function. The RSB reviewer, with the CPB and EICSB reviewers, ascertains that the reactivity and response characteristics of the reactivity control system are conservative with respect to the parameters assumed in the Chapter 15 analyses. In the Chapter 15 review, the RSB reviewer verifies that no credit has been taken for the RFCS (in BWR's) to mitigate any accident. (Although the

RFCS controls reactor power level over a limited range, it is not required for shutdown.) In addition, the reviewer reviews the operation of the RFCS to confirm that a malfunction or failure of the system will not degrade the capabilities of plant safety systems or lead to plant conditions more severe than those considered in the accident analyses (e.g., by determining the effects of a failure of the system following a loss-of-coolant accident or steam line break). The RSB, in SRP 15.4.5, reviews the results of the most limiting transient from a malfunction of the RFCS.

IV. EVALUATIONS FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The functional designs of the reactivity control systems for the _____ plant have been reviewed to confirm that the systems have the capability to shut down the reactor with appropriate margin during normal, abnormal and accident conditions. The reactivity control systems reviewed included the CRDS and _____ (CVCS for PWR's or SLCS and RFCS for BWR's). The scope of review included process flow diagrams, layout drawings, piping and instrumentation diagrams, and descriptive information for the systems and for the supporting systems that are essential for operation of the systems. [The applicant's proposed design criteria and design bases for the reactivity control systems and the adequacy of those criteria and bases have been reviewed. (CP)] [The manner in which the design of the reactivity control systems and supporting systems conform to the proposed design criteria and bases has been reviewed. (OL)]

"The basis for staff acceptance has been conformance of the applicant's designs, design criteria, and design bases for the reactivity control systems and their supporting systems to the Commission's regulations as set forth in the general design criteria of 10 CFR Part 50.

"The staff concludes that the designs of the reactivity control systems conform to all applicable regulations and are acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 5.2.1.1* COMPLIANCE WITH 10 CFR § 50.55a

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

In order to establish that pressure-retaining components of the reactor coolant pressure boundary of water-cooled nuclear power plants are in compliance with 10 CFR § 50.55a, an applicant is required to provide a table in his safety analysis report (SAR) identifying pressure vessels, piping, pumps and valves and the component code, code edition, applicable addenda, and component order date (where applicable) of each such component. Pressure-retaining components of the reactor coolant pressure boundary are designated as Class I components under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter the Code). 10 CFR § 50.55a requires that these components meet the requirements for Class I components under the Code.

For construction permit (CP) and operating license (OL) applications, the ASB will determine the acceptability of the information presented in the SAR, to assure that the applicant is in compliance with the rules of Section 50.55a.

In the event there are cases where conformance to Section 50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, the applicant must provide a complete description of the circumstances and the basis for proposed alternate requirements. The applicant must describe how an equivalent and acceptable level of safety and quality will be provided by the proposed alternate requirements.

When required, the MEB and MTEB will provide assistance in establishing acceptability in the event an applicant invokes the "hardship" clause, and does not conform in all respects with Section 50.55a.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 1. This criterion requires that structures, systems, and components important to safety shall be designed, fabricated,

*Formally 5.2.1.3

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erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR § 50.55a. This rule establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components within the reactor coolant pressure boundary of boiling and pressurized water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

3. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

III. REVIEW PROCEDURES

The table provided by the applicant identifying pressure vessel components, piping, pumps and valves, and the corresponding component code, code edition, applicable addenda, and, when required, the component order date of each Code Section III, Class 1 component within the reactor coolant pressure boundary, is checked for compliance with Section 50.55a of 10 CFR Part 50. This review is applicable to CP and OL applications.

For those components within the reactor coolant pressure boundary not in compliance with the rules of Section 50.55a, a review of the code and code addenda is performed, to identify the specific sections with which the component does not comply. A decision to accept a component which is not fully in compliance with the rules is based on a judgment of the relative importance of the specific provisions in the code or code addenda not complied with, and a determination that any noncompliance will not result in an unacceptable level of safety and quality.

If the staff's concerns are not resolved in a satisfactory manner, a staff position is taken requiring conformance with the rules of Section 50.55a.

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information is contained in the SAR and amendments and that his evaluation supports conclusions of the following type, which are to be included in the staff's safety evaluation report:

"The components of the reactor coolant pressure boundary, as defined by the rules of 10 CFR § 50.55a, have been properly identified and classified as ASME Code Section III Class I components. These components will be constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of 10 CFR § 50.55a.

"The staff concludes that construction of the components of the reactor coolant pressure boundary in conformance with these codes provides assurance that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. 10 CFR § 50.55a, "Codes and Standards Rule."

4. ASME Boiler and Pressure Vessel Code, 1977 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers (1977).

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SECTION 5.2.1.2* APPLICABLE CODE CASES

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The ASB determines the acceptability of American Society of Mechanical Engineers (ASME) and American National Standards Institute (ANSI) code case interpretations specified in the safety analysis report (SAR). These code cases must be approved before being applied to ASME Boiler and Pressure Vessel Code, Section III, Class 1 pressure-retaining components within the reactor coolant pressure boundary, as stated in the Codes and Standards Rule, Section 50.55a(a)(2)(ii) of 10 CFR Part 50. These code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A.

The MEB and MTEB, on a generic basis, determine the acceptability of ASME and ANSI code case interpretations that may be applied to ASME Code Section III, Class 1 pressure-retaining components within the reactor coolant pressure boundary (Quality Group Classification A). These branches review each revision to applicable code cases. Code cases pertaining to materials, fabrication, and nondestructive testing are evaluated by the MTEB. All other areas covered by ASME code cases are evaluated by the MEB.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 1. This criterion requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR § 50.55a. This rule establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components within the reactor coolant pressure boundary of boiling and pressurized water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

*Formally 5.2.1.4

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3. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

4. Regulatory Guide 1.84, "Code Case Acceptability in ASME Section III - Design and Fabrication." This guide lists those Section III ASME code cases oriented to design and fabrication which are acceptable to the staff for implementation in the licensing of nuclear power plants.

5. Regulatory Guide 1.85, "Code Case Acceptability in ASME Section III - Materials." This guide lists those Section III ASME code cases oriented to materials and testing which are acceptable to the staff for implementation in the licensing of nuclear power plants.

II. REVIEW PROCEDURES

The table provided by the applicant identifying those ASME code cases applied to Section III, Class 1 pressure-retaining components within the reactor coolant pressure boundary is checked for compliance with the list of acceptable code cases identified in Regulatory Guides 1.84 and 1.85 and Tables 5.2.1.2-1 and 5.2.1.2-2.

Code cases applicable to other sections of the ASME Boiler and Pressure Vessel Code and referenced by Section III, Division 1, are also reviewed to determine their acceptability.

Table 5.2.1.2-1 lists those ASME Sections V and IX Code Cases oriented to nondestructive examination and welding and brazing qualifications, which are acceptable to the staff for implementation in the licensing of nuclear power plants.

Table 5.2.1.2-2 lists those ASME Section XI Code Cases oriented to inservice inspection which are acceptable to the staff for implementation in the licensing of nuclear power plants.

ASME Section III, Division 2, Code Cases oriented to Concrete Reactor Vessels and Containments are reviewed by the SEB on a case-by-case basis for implementation in the licensing of nuclear power plants pending approval of Section III, Division 2 of the code and the associated code cases by the Commission.

In the event an applicant should propose the use of a code case not previously reviewed by the staff, a review of the code case is requested of the MEB or MTEB, as appropriate.

IV. EVALUATION FINDINGS

The staff review should verify that only acceptable ASME and ANSI code cases are specified in the SAR in order to arrive at conclusions of the following type, which are to be included in the staff's safety evaluation report:

"The specified ASME and ANSI code cases whose requirements will be applied in the construction of pressure-retaining ASME Code Section III, Class 1 components within the reactor coolant pressure boundary (Quality Group Classification A), are acceptable to the staff. The staff concludes that compliance with the requirements of these code cases will result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. 10 CFR § 50.55a, "Codes and Standards Rule."

4. ASME Boiler and Pressure Vessel Code, 1977 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers (1977).

5. ASME Boiler and Pressure Vessel Code, 1977 Edition, "Code Cases, Boilers and Pressure Vessels," American Society of Mechanical Engineers (1977).

6. ASME Boiler and Pressure Vessel Code, 1977 Edition, "Code Cases, Nuclear Components," American Society of Mechanical Engineers (1977).

7. Regulatory Guide 1.84, "Code Case Acceptability in ASME Section III - Design and Fabrication."

8. Regulatory Guide 1.85, "Code Case Acceptability in ASME Section III - Materials."

TABLE 5.2.1.2-1 CODE CASE ACCEPTABILITY, ASME SECTION V-NONDESTRUCTIVE EXAMINATION, AND ASME SECTION IX WELDING AND BRAZING QUALIFICATIONS

Code Case	Date of Council Approval	Title
1400	9-20-68	Welder Performance Qualification Tests for Joints of Non-ferrous Material, Section IX
1452	2-27-70	Use of Brazing Filler Material BAg-2a, Section IX
1632	3-2-74	Referencing the 5 Percent Nickel (P-11A Subgroup 2) Materials in Par. Q-10(b) of Section IX in Addition to the 9 Percent Nickel (P-11A Subgroup 1) Materials.
1693	8-11-75	Welding Procedure Qualification of Dissimilar Metal Welds when "Buttering" with Alloy Weld Metal and Heat Treatment May be Involved, Section IX
N-92 (1968)	6-30-75	Waiver of Ultrasonic Transfer Method, Sections III, V and VIII, Division 1. Code Case 1698 is acceptable subject to the following conditions in addition to those conditions specified in the Code Case. The method used to demonstrate actual correlation of the basic calibration block with production testing should be demonstrated and justified for each UT application.
1707	11-30-75	Alternate Performance Qualifications, Section IX (Applicable to all Sections, Including Section III, Division 1)
1816	3-23-77	Use of Immersion Technique in Ultrasonic Examination per SA-388, Sections I, II, V, and VII, Divisions 1 and 2

TABLE 5.2.1.2-2 CODE CASE ACCEPTABILITY, ASME SECTION XI-INSERVICE INSPECTION OF NUCLEAR POWER PLANT COMPONENTS

Code Case	Date of Council Approval	Title
N-34 (1551)	11-6-72	Inservice Inspection of Welds on Nuclear Components, Section XI
N-72 (1646)	8-12-74	Partial Postponement of Section XI Category B-C Examinations for Class 1 Components
N-73 (1647)	8-12-74	Partial Postponement of Section XI Category B-D Examinations for Class 1 Components
N-98 (1705-1)	3-1-76	Ultrasonic Examination - Calibration Block Tolerances, Section XI
N-112 (1730)	11-3-75	Acceptance Standards for Section XI, Division 1, Class 2 and 3 Components
N-113 (1731)	11-3-75	Basic Calibration Blocks for Section XI, Division 1 - Ultrasonic Examination of Welds 10 in. to 14 in. Thick
N-118 (1738)	12-2-75	Acceptance Standards for Surface Indications in Cladding, Section XI
N-167 (1804)	1-14-77	Minimum Section Thickness Requirements for Repair of Nozzles, Section XI, Division 1

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SECTION 5.2.2 OVERPRESSURIZATION PROTECTION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Overpressure protection for the reactor coolant pressure boundary (RCPB) is provided by means of relief and safety valves. For RCPB overpressure protection, the relief and safety valves operate in conjunction with the reactor protection system and the steam generator safety valves. The reviewer examines the design bases, system and component descriptions, and system analyses and tests described in the applicant's safety analysis report (SAR) in order to evaluate the adequacy of the overpressure protection which is provided. The areas of review for a boiling water reactor (BWR) are the reactor coolant system relief and safety valves. For a pressurized water reactor (PWR), the areas of review are the pressurizer safety and relief valves, and the piping from these valves to the quench tank. The review of anticipated transients without scram is described in Standard Review Plan (SRP) 15.8.

The adequacy of the proposed preoperational and initial startup test programs is examined as a part of this review. The reviewer also evaluates the proposed technical specifications to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The overpressure protection components are also reviewed to assure that they have the proper seismic and quality group classification. This aspect of the review is performed as a portion of the effort described in SRP 3.2.1 and SRP 3.2.2.

The MEB, as described in SRP 3.9.3, reviews the design and installation criteria for the overpressure protection components to assure that they are in conformance with ASME Boiler and Pressure Vessel Code requirements.

The EICSB, as described in SRP 7.6, evaluates the adequacy of controls and instrumentation of the overpressure protection components with regard to the required features of automatic actuation, remote sensing and indication, remote control, emergency onsite power, and connections to the reactor protection system.

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The CPB provides generic evaluation of the mathematical models used to analyze transients that result in an increase in the pressure within the reactor coolant system.

The APCSB reviews the adequacy of the pressure relief and safety valves for the secondary system of PWR's.

II. ACCEPTANCE CRITERIA

The fundamental criterion against which an evaluation of overpressure protection is to be made is General Design Criterion 15 (Ref. 1): "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." Further, the preoperational and initial startup test programs are to meet the intent of Regulatory Guide 1.68 (Ref. 2).

To be acceptable, adequate relief and safety valve capacity must be provided for the primary systems of PWR's and BWR's. For PWR's, the secondary system must also be provided with relief and safety valves having adequate capacity.

1. Relief Valves

For the design basis normal operational transients, the relief valve capacity must be sufficient to limit the pressure so as to prevent safety valve discharge directly to the containment, with the following assumptions:

a. The reactor is operating at licensed core thermal power level.

b. All system and core parameters are at the values within the normal operating ranges which would produce the highest transient pressure.

c. All components, instrumentation, and controls function normally.

2. Safety Valves

For the most severe abnormal operational transient, with reactor scram, the safety valve capacity should be sufficient to limit the pressure to less than 110% of the RCPB design pressure, as specified by the ASME Boiler and Pressure Vessel Code (Ref. 3), with sufficient margin to account for uncertainties in the design and operation of the plant and assuming:

a. The reactor is operating at a power equal to the licensed core thermal power level plus an increment sufficient to account for power measurement uncertainties.

b. All system and core parameters are at the values within the normal operating range, including uncertainties and technical specification limits, which would result in the highest transient pressure.

c. The reactor scram is initiated either by the high pressure signal or by the second signal from the reactor protection system, whichever is later.

d. The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code for each type of valve.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For operating license (OL) applications, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report and in the report on overpressure protection. The latter report is required by the ASME Code and is to serve as the basis for many of individual review steps outlined below during the OL review. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The following steps are taken by the RSB reviewer in determining that the acceptance criteria of Section II have been met. These steps should be applied to CP and OL reviews as appropriate. Previously reviewed designs may be used as a guide; however, the reviewer must verify that any changes are justified.

1. The piping and instrumentation diagrams are examined to determine the number, type, and location of the safety and relief valves in both the primary and secondary systems, and of discharge lines, instrumentation, and other components.

2. All other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems are identified. The effects of these other functions or systems on operation of the overpressure protection system are determined.

3. The valve descriptions are examined to determine type and manufacturer and to evaluate reliability (e.g., new or standard design).

4. The capacities, set points, and setpoint tolerances for all safety and relief valves are identified.

5. All of the reactor trip signals which occur during overpressure transients, including their setpoints and setpoint tolerances, are identified.

6. All transients analyzed in Chapter 15 of the SAR that result in an increase in the pressure experienced by the RCPB are examined. The peak predicted pressures are identified and the operating conditions and setpoints used in the analysis are reviewed to assure that they are suitably conservative.

The information below is provided to the reviewer as guidance and is based on typical previously-reviewed designs.

a. BWR's - For relief valve sizing, in transients in which a scram is initiated by closure of the main steam isolation valves or fast closure of the turbine stop valves, the highest pressure results from instantaneous loss of condenser vacuum or a turbine trip without bypass. For safety valve sizing, in transients in which scram is initiated by high flux or high pressure, the highest pressure results from closure of all main steam isolation valves. Analysis of previously acceptable designs has shown that the peak pressure is at least 25 psi below the allowable, assuming the reactor is operating at 105% of rated power, pressure is 1040 psia, no credit is taken for relief valve capacity, one safety valve fails to open, and scram is initiated by high pressure.

b. PWR's - For relief valve sizing, the valve capacity has been sufficient to accommodate the surge from the design basis step load change. Safety valve sizing is usually based on the maximum surge rate that results from a turbine trip without bypass. Analysis of previously acceptable designs has shown that the discharge flow from the safety valves in the primary and secondary systems is typically 86% and 100% of their respective rated capacities assuming the reactor is initially at 102% of rated power, the uncertainties in power, pressure, and temperature are 5%, 30 psi, and 4°F, respectively, scram is initiated by low level in the steam generator; and no credit is taken for relief valve operation or Doppler or moderator temperature reactivity feedback.

". The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 2). At the OL stage, this aspect of the review is to assure that sufficient information is provided by the applicant to identify clearly the test objectives, methods of testing, and acceptance criteria (See par. C.2.b of Regulatory Guide 1.68.)

The reviewer evaluates the proposed test programs to determine if they provide a reasonable assurance that the components that provide overpressure protection will perform their safety function. As an alternative to this detailed evaluation, the reviewer may compare the overpressure protection design to that of a previously reviewed plant. If the design is essentially identical and if the proposed test programs are essentially the same as performed previously on other plants, the reviewer may conclude that the proposed test programs for overpressure protection are adequate.

If the proposed design differs significantly from that of previously reviewed designs, the impact of the proposed changes on the preoperational and initial startup testing programs are reviewed at the construction permit stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

8. The proposed plant technical specifications are reviewed to:

   a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

   b. Verify that the frequency and scope of periodic surveillance testing is adequate.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

1. BWR's

"The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressure protection will be provided by _____ safety and relief valves located on the four main steam lines between the reactor vessel and the first isolation valve inside the drywell. The relief and safety valves are distributed among the four main steam lines such that a single accident cannot disable the safety, relief, or automatic depressurization functions. The valves discharge through piping to the suppression pool. The valves operate as spring-loaded safety valves with set pressures that range from _____ to _____ psig. Their total capacity at their set pressure is _____% of rated steam flow.

"To determine the ability of the pressure relief system to prevent overpressurization, the applicant analyzed the severe transient of main steam isolation valve closure. The analysis was performed assuming that: a) the plant is in operation at design conditions (*% of rated steam flow and a reactor vessel dome pressure of * psig), and b) the reactor is shut down by a high pressure scram. The calculated peak pressure at the bottom of the vessel is _____ psig, a margin of _____ psi below the code allowable of _____ psig (110% of vessel design pressure). The staff concludes that the design of the pressure relief systems conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards and is acceptable."

2. PWR's

"The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressure protection for the reactor coolant pressure boundary is accomplished by utilizing the _____ safety valves. These valves discharge to the pressurizer quench tank through a common header from the pressurizer. The reactor coolant system (RCS) safety valves, in conjunction with the steam generator safety valves, and the reactor protection system, will protect the RCS against overpressure in the event of a complete loss of heat sink.

*Normally, BWR's are analyzed at 105% rated steam flow at a pressure of 1040 psig.

"The peak RCS pressure following the worst transient is limited to the ASME Code allowable (110% of the design pressure) with no credit taken for operation of RCS relief valves, steam line relief valves, steam dump system, RCS pressurizer level control system, or pressurizer spray. The _____ plant was assumed to be operating at design conditions (_% of rated power) and the reactor is shut down by a _____ scram. The calculated pressure at the bottom of the vessel is _____ psig, a margin of _____ psi below the code allowable of _____ psig (110% of vessel design pressure).

The staff concludes that the design of the pressure relief system conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

2. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

3. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

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SECTION 5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 1, 13, 14 and 31 require that the reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of a rapidly propagating failure and of a gross rupture. The Criteria also require that the reactor coolant pressure boundary shall be tested to quality standards commensurate with the safety function to be performed and that instrumentation shall be provided to monitor the variables that can affect the integrity of the reactor coolant pressure boundary.

The following areas, which relate to materials of the reactor coolant pressure boundary (RCPB) other than the reactor pressure vessel, which is covered in Standard Review Plan 5.3.1, "Reactor Vessel Materials", are reviewed.

1. Material Specifications

The specifications for pressure-retaining ferritic materials and austenitic stainless steels, including weld materials, that are used for each component (e.g., vessels, piping, pumps, and valves) of the reactor coolant pressure boundary, are reviewed.

The adequacy and suitability of the ferritic materials, stainless steel, and nonferrous metals specified for the above applications are reviewed.

2. Compatibility of Materials with the Reactor Coolant

General corrosion and stress corrosion cracking induced by impurities in the reactor coolant can cause failures of the reactor coolant pressure boundary.

The chemistry of the reactor coolant and the additives (such as inhibitors) whose function is to control corrosion are reviewed. The water chemistry includes the permissible concentrations of chlorides, fluorides, oxygen, hydrogen, and soluble poisons, the methods used to control the concentrations of impurities, and the pH.

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The review includes the compatibility of the materials of construction employed in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed. The extent of the corrosion of ferritic low alloy steels and carbon steels in contact with the reactor coolant is reviewed. Similarly, a review is made of possible uses of austenitic stainless steels in the sensitized condition. The use of austenitic stainless steels in boiling water reactors (BWR's) requires special attention because of the oxygen content of BWR coolant.

3. Fabrication and Processing of Ferritic Materials

a. The fracture toughness properties of ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary are reviewed.

The fracture toughness tests performed on all ferritic materials used for pressure-retaining RCPB components (i.e., vessels, pumps, valves, and piping) are reviewed.

The test procedures used for Charpy V-notch impact and dropweight testing are reviewed.

Fracture toughness of the material is characterized by its reference temperature, RT_{NDT}. This temperature is the higher of the nil-ductility temperature (NDT) from the dropweight test or the temperature that is 60°F below the temperature at which Charpy V-notch impact test data are 50 ft-lb and 35 mils lateral expansion. The limiting RT_{NDT} temperature of the material is reviewed.

b. The control of welding in ferritic steels is reviewed.

(1) The quality of welds in low alloy steels can be increased significantly by proper controls. In particular, the propensity for cold cracks or reheat cracks to form in areas under the bead and in heat-affected zones (HAZ) can be minimized by maintaining proper preheat temperatures of the base metal concurrent with controls on other welding variables. The minimum preheat temperature and the maximum interpass temperatures are reviewed.

(2) The quality of electroslag welds in low alloy steel components can be increased by maintaining a weld solidification pattern that possesses a strong intergranular bond in the center of the weld. The welding variables, which have a significant effect on the weld solidification pattern, must be controlled. The welding variables, solidification patterns, macroetch tests, and Charpy V-notch impact tests of electroslag welds are reviewed.

(3) Experience shows that a welder qualified to weld low-alloy steel or carbon steel components under normal fabricating conditions may not produce acceptable welds if the accessibility to the weld area is restricted.

Limited accessibility can occur when component parts are joined in the final assembly or at the plant site, where other adjacent components or structures prevent the welder from assuming an advantageous position during the welding operation. The adequacy of accessibility during the welding of ferritic components is reviewed.

c. The requirements for nondestructive examination of ferritic wrought seamless tubular products used for ASME Class 1 components of nuclear power plants are specified in Paragraphs NB-2550 through NB-2570, ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III. The methods of examination specified for nondestructive examination are reviewed.

4. Fabrication and Processing of Austenitic Stainless Steel

Austenitic stainless steels in a variety of product forms are used for construction of pressure-retaining components in the reactor coolant pressure boundary. Unstabilized austenitic type stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316, are normally used. Because these compositions are susceptible to stress corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress corrosion cracking.

a. Sensitization is caused by intergranular precipitation of chromium carbide in austenitic stainless steels that are exposed to temperatures in the approximate range of 800°F to 1500°F. Precipitation increases with increasing carbon content and exposure time. Control of the application and processing of stainless steel is needed to eliminate the occurrences of stress corrosion cracking in sensitized stainless steel components of nuclear reactors. Test data and service experience demonstrate that sensitized stainless steel is significantly more susceptible to stress corrosion cracking than nonsensitized (solution heat treated) stainless steel.

Special provisions may apply to the use of austenitic stainless steel in boiling water reactor (BWR) piping because plant operating experience indicates that reactor coolant boundary piping is susceptible to oxygen-assisted stress corrosion cracking.

b. The following areas are reviewed: requirements for solution heat treatment of stainless steel; plans to avoid partial or severe sensitization during welding, including information on welding methods, heat input, and interpass temperatures; and a description of the material inspection program that will be used to verify that unstabilized austenitic stainless steels are not susceptible in service to intergranular attack.

Contamination of austenitic stainless steel with halogens and halogen-bearing compounds (e.g., die lubricants, marking compounds, and masking tape) must be avoided to the maximum degree possible to avoid stress corrosion cracking. Plans for cleaning and protecting the material against contaminants capable of causing stress corrosion cracking during fabrication, shipment, storage, construction, testing, and operation of components and systems are reviewed. Any pickling used in processing austenitic stainless steel components and the restrictions placed on pickling sensitized materials are reviewed. The upper limit on the yield strength of austenitic stainless steel materials is reviewed.

c. Whether sensitized or not, austenitic stainless steel is subject to stress corrosion and must be protected from contaminants that can promote cracking. Thermal insulation is often employed adjacent to, or in direct contact with, stainless steel piping and components. The contaminants present in the thermal insulation may be leached by spilled or leaking liquids and deposited on the stainless steel surfaces. The controls on the use of nonmetallic thermal insulation are reviewed.

d. Austenitic stainless steel is subject to hot cracking (microfissuring) during welding if the weld metal composition or the welding procedure is not properly controlled. Because cracks formed in this manner are small and difficult to detect by nondestructive testing methods, welding procedures, weld metal compositions, and delta ferrite percentages that minimize the possibility of hot cracking must be specified. As a part of achieving this control, Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" contains recommendations for process control through the testing of weld test pads. The staff recommendations will provide assurance that the ferrite content will be adequate to prevent microfissuring. The adequacy of the proposed welding procedures is reviewed.

The assurance of satisfactory electroslag welds for austenitic stainless steel components can be increased by maintaining a weld solidification pattern with a strong intergranular bond in the center of the weld. The welding variables that have a significant effect on the weld solidification pattern must be controlled.

A number of electroslag welding process variables, such as, slag pool depth, electrode feed rate and oscillation, current, voltage, and slag conductivity, have been shown to influence the weld solidification pattern. If the combination of process variables produces a deep pool of molten weld metal, the crystal (dendritic) growth direction from the pool sides will join at an obtuse angle at the center of the weld, and cracks may develop because of the weaker centerline bond between dendrites. A proper combination of process variables promotes a dendritic growth pattern with an acute joining angle, which results in a strong

centerline bond. The welding variables, solidification patterns, and macro-etch tests used in the electroslag welding of austenitic stainless steel are reviewed.

Experience has shown that a welder qualified to weld stainless steel components under normal fabricating conditions may not produce acceptable welds if the accessibility to the weld area is restricted. Limited accessibility can occur when component parts are joined in the final assembly or at the plant site, where other adjacent components or structures prevent the welder from assuming an advantageous position during the welding operation. The adequacy of accessibility of field erected structures, for welding austenitic stainless steel components, is reviewed.

e. The requirements for nondestructive examination of wrought seamless tubular products used for components of nuclear power plants are specified in Paragraph NB-2550 of the Code, Section III. Nondestructive examination techniques applied to tubular products used for components of the RCPB, or other safety-related ASME Class 1 systems that are designed for pressure in excess of 275 psig or temperatures in excess of 200°F, must be capable of detecting unacceptable defects regardless of defect shape, orientation, or location in the product.

The nondestructive examination procedures used for inspection of tubular products are reviewed.

Inservice inspection requirements for the Reactor Coolant Pressure Boundary are described in SRP 5.2.4, "Inservice Inspection and Testing of Reactor Coolant Pressure Boundary."

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Material Specifications

The specifications for permitted materials are those identified in the ASME Code, Section III, Appendix I, or described in detail in the ASME Code, Section II, Parts A, B, and C. Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials," describes the acceptable Code Cases to be used in conjunction with the above specifications.

Special requirements for BWR piping materials and materials processing are described in Branch Technical Position MTEB 5-7,* "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

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* Branch Technical Position MTEB 5-7 is identical to the content of NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

2. Compatibility of Materials with the Reactor Coolant

In boiling water reactors (BWR's), high purity water is maintained. The purity is monitored through continuous on-line reading of the conductivity of the coolant and by periodically sampling and chemically analyzing it for pH and chloride content. An on-line water treatment plant maintains the coolant within Technical Specification limits. In reactor coolants used for BWR's, oxygen seeks a natural level, and no attempt is made to control the amount of oxygen contained in the solution. The acceptance criteria for the chemistry of the BWR coolant are specified in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors." Acceptable locations for chemistry monitoring sensors are described in Regulatory Guide 1.56.

In reactor coolants used for pressurized water reactors (PWR's), the conductivity measurements tend to show high values due to interference from additions of boric acid. These additions tend to mask the effect of other impurities. Therefore, sampling and chemical analysis for chlorides, fluorides, and oxygen must be performed on a scheduled basis. Acceptable levels of chloride, fluoride and oxygen for pressurized water reactor coolant purity are stated in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

Ferritic low alloy steels and carbon steels, which are used in many principal pressure-retaining components, are clad with a layer of austenitic stainless steel. If cladding is not used, conservative corrosion allowances must be indicated for all exposed surfaces of carbon and low alloy steels, as indicated in the ASME Code, Section III, NB-3120, "Corrosion."

Unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB must conform to requirements of Regulatory Guide No. 1.44 and Branch Technical Position MTEB 5-7, including verification of nonsensitization of the material by an approved test.

3. Fabrication and Processing of Ferritic Materials

a. The acceptance criteria for fracture toughness are provided by the ASME Code, Section III; and 10 CFR Part 50, Appendix G.

The pressure-retaining components of the RCPB that are made of ferritic materials must meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. With respect to absorbed energy in ft-lbs and lateral expansion as shown by Charpy V-notch (C_v) impact tests, all materials must meet the acceptance standards of Article NB-2330 of the Code, Section III, and the requirements of Sections IV.A.2, IV.A.3, and IV.B of Appendix G, 10 CFR Part 50, as follows:

(1) The special acceptance requirements for fracture toughness of reactor vessels are covered by Standard Review Plan 5.3.1, "Reactor Vessel Materials."

(2) Materials for piping (i.e., pipes, tubes, and fittings), pumps, and valves, excluding bolting materials, must meet the requirements of the Code, Section III; Paragraph NB-2332, and Appendix G, Paragraph G-3100. The required C_v values for piping are specified in Table NB-2332-1 of the Code, Section III.

(3) Materials for bolting for which impact tests are required must meet the requirements of the Code, Section III, Paragraph NB-2333, and Appendix G, Paragraph G-4100

(4) Calibration of instruments and equipment must meet the requirements of the Code, Section III, Paragraph NB-2360.

b. The acceptance criteria for control of ferritic steel welding are listed below:

(1) The amount of specified preheat must be in accordance with the requirements of the Code, Section III, Appendix D, Paragraph D-1200, supplemented by Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steel."

The supplemental acceptance criteria for control of preheat temperature are as follows:

The welding procedure qualification requires that minimum preheat and maximum interpass temperatures be specified and that the welding procedure be qualified at the minimum preheat temperature. For production welds, the preheat temperature should be maintained until a post-weld heat treatment has been performed.

The preheat controls described in the Westinghouse Topical Report WCAP-8577 are an acceptable alternate to compliance with those of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steel."

Production welding should be monitored to verify that the limits on preheat and interpass temperatures are maintained. In the event that the above criteria are not met, the weld is subject to rejection.

(2) The acceptance criteria for electroslag welds are presented in Regulatory Guide 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns and impact test limits (for qualification of welds in Class 1 and Class 2 components) and the criteria for verifying conformance during production welding.

(3) Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for requalification of welders: the performance qualification should require testing of the welder when

conditions of accessibility to a production weld are less than 30 to 35 cm (12-14 inches) in any direction from the joint; and requalification is required for different restricted accessibility conditions or when any of the essential variables listed in the Code, Section IX, are changed. Qualification of the Welder or Welding Operators for limited accessibility may be waived provided that 100% Radiographic and/or Ultrasonic examination of the completed welded joint is performed. Examination procedures and acceptance standards should meet the requirements of the ASME Section III of the Code. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance Documentation for the completed weld.

c. Acceptance criteria for nondestructive examination of ferritic steel tubular products are given in the ASME Code, Section III, Paragraph NB-2550.

4. Fabrication and Processing of Austenitic Stainless Steel

a. The acceptance criteria for testing, alloy compositions, and heat treatment, to avoid sensitization in austenitic stainless steels, are covered in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and additional criteria for BWR's are in Branch Technical Position MTEB 5-7.

b. Controls to avoid stress corrosion cracking in austenitic stainless steels are also covered in Regulatory Guide 1.44. This guide provides acceptance criteria on the cleaning and protection of the material against contaminants capable of causing stress corrosion cracking. Acid pickling is to be avoided on fabricated stainless steels. Necessary pickling is to be done only with appropriate controls. Pickling should not be performed upon sensitized stainless steels.

The quality of water used for final cleaning or flushing of finished surfaces during installation is in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." Vented tanks with deionized or demineralized water are an acceptable source of water for final cleaning or flushing of finished surfaces. The oxygen content of the water need not be controlled.

Laboratory stress corrosion tests and service experience provide the basis for the criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit on the yield strength of 90,000 psi.

c. The compatibility of austenitic stainless steel materials with thermal insulation is dependent upon the type of insulation. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation which meets the criteria of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation

for Austenitic Stainless Steel" is used. The acceptance criteria for nonmetallic insulation for stainless steel are based on the levels of leachable contaminants in the material and are presented in position C.2.b and Figure 1 of the guide.

d. The acceptance criteria for delta ferrite in austenitic stainless steel welds are given in Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." These acceptance criteria cover (1) verification of delta ferrite content of filler metals, (2) ferrite measurement, (3) instrumentation, (4) acceptability of test results, and (5) documentation of weld pad verification test.

The acceptance criteria for electroslag welds in austenitic stainless steel are given in Regulatory Guide 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns for qualification of austenitic stainless steel welds and the basis for verifying conformance during production welding.

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for requalification of welders:

(1) The performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cm (12-14 inches) in any direction from the joint.

(2) Requalification is required for different restricted accessibility conditions or when other essential variables listed in the Code, Section IX, are changed. An alternate acceptance criterion is as stated in II.3.b.

e. The acceptance criteria for nondestructive examination of austenitic stainless steel tubular products are stated in the ASME Code, Section III, Paragraph NB-2550.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review described in Section I of this plan, the following review procedures are followed.

1. Material Specifications

The material specifications for each major pressure-retaining component or part used in the RCPB are compared with the acceptable specifications listed in the Code, Sections II and III, as stated in the acceptance criteria. Exceptions to the material specifications of the Code are clearly identified, and the basis evaluated. The reviewer judges the significance of the exceptions and, taking into account precedents set in earlier cases, determines the acceptability of the proposed exceptions. In those instances where the Materials Engineering Branch takes exception to the use of

a specific material or questions certain aspects of a specification, the applicant is advised which material is not acceptable, and for what reason.

2. Compatibility of Materials with the Reactor Coolant

The reviewer verifies that the following information is provided at each respective stage of the review process:

a. At the construction permit stage of review:

(1) A list of the materials of construction of the components of the reactor coolant pressure boundary that are exposed to the reactor coolant, including a description of material compatibility with the coolant, contaminants, and radiolytic products to which the materials may be exposed in service.

(2) A list of the materials of construction of the RCPB, and a description of material compatibility with external insulation and with the environment in the event of reactor coolant leakage.

(3) The limitations imposed on concentrations of chloride and fluoride ions and oxygen in the reactor coolant, and the extent of monitoring such limitations.

(4) The fabrication and cleaning controls imposed on stainless steel components to minimize contamination with chloride and fluoride ions.

(5) The controls and limits that are specified for leachable impurities in thermal insulation, as identified in Section II.4.c.

(6) For BWR's, performance monitoring recommended in Regulatory Guide No. 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

b. At the operating license stage of the review process:

(1) The items listed under 2.a above, to provide assurance that any changes are noted that may have occurred during the period between the submittal of SAR's.

(2) A list of the instrumentation and equipment that will monitor and control the purity of the reactor coolant, including water purity indicators and alarms provided in the control room.

3. Fabrication and Processing of Ferritic Materials

a. The information submitted by the applicant relative to tests for fracture toughness is reviewed for conformance with the acceptance criteria stated In Section II.3.a. These tests include Charpy V-notch impact and dropweight tests. A description of the tests is reviewed, and the locations of the test specimens and their orientation are verified. Information regarding calibration of instruments and equipment is reviewed for conformance with the acceptance criteria stated in Section II.3.a.(4).

In the event that none of the fracture toughness tests has been performed, the preliminary safety analysis report (PSAR) must contain a statement of the applicant's intention to perform this work in accordance with the Code, Section III, Paragraph NB-2300 and Appendix G; and the requirements of 10 CFR 50, Appendix G.

The final safety analysis report (FSAR) is reviewed to assure that all the impact tests required by NB-2340 have been performed.

b. The control of welding in ferritic steels is reviewed as described below:

(1) The information submitted by the applicant regarding the control of preheat temperatures for welding low alloy steel is reviewed for conformance with the acceptance criteria stated in Section II.3.b.(1).

(2) The electroslag weld information submitted by the applicant is reviewed for conformance to the acceptance criteria discussed in Section II.3.b.(2). The information in the SAR is reviewed to verify that macroetch tests have been made (to assure that an acceptable weld solidification pattern is obtained) and that impact tests specified in Regulatory Guide 1.34 meet the acceptance criteria discussed previously in Section II.3.b.(2).

(3) The ASME Code, Section III, requires adherence to the requirements of Section IX, "Welding Qualifications." One of the requirements is welder qualification for production welds. However, there is a need for supplementing this section of the Code because the assurance of providing satisfactory welds in locations of restricted direct physical and visual accessibility can be increased significantly by qualifying the welder under conditions simulating the space limitations under which the actual welds will be made.

Regulatory Guide 1.71, "Welder Qualification for Limited Accessibility," provides the necessary supplement to the Code, Section IX, in this respect. The information submitted by the applicant is reviewed for conformance with acceptance criteria discussed in Section II.3.b.(3).

c. The ASME Code, Section III, NB-2550 specifies the ultrasonic method for examination of ferritic steel tubular products.

4. Fabrication and Processing of Austenitic Stainless Steels

a. The information submitted by the applicant in the following areas is reviewed for conformance with the acceptance criteria stated in Section II.4.a regarding:

(1) The desirable stage in the sequence of processing for solution heat treatment, the rates of cooling, and the quenching media.

(2) Controls to prevent sensitization during welding, as described in Regulatory Guide 1.44.

(3) Controls to verify non-sensitization, as described in Regulatory Guide 1.44.

(4) For BWR's, additional processing controls, as described in Branch Technical Position MTEB 5-7.

In the event that information in the above areas is not supplied, sufficient justification for the deviation must be presented.

b. The information submitted by the applicant is reviewed for conformance with the acceptance criteria discussed in Section II.4.b as follows:

Verification is sought that process controls are exercised during all stages of component manufacture and reactor construction to minimize the exposure of austenitic stainless steels to contaminants that could lead to stress corrosion cracking.

Information is also checked to assure that precautions have been taken to require removal of all cleaning solutions, processing compounds, degreasing agents, and any other foreign material from the surfaces of the component at any stage of processing prior to any elevated temperature treatment and prior to hydrotests. The reviewer verifies that a statement is contained in the SAR that pickling of sensitized austenitic stainless is avoided and that the quality of water used for final cleaning or flushing of finished surfaces during installation is in accordance with acceptance criteria discussed in Section II.4.b.

Because excessive cold work in austenitic stainless steel can render this material susceptible to stress corrosion cracking, control must be exerted by the applicant, by placing an upper limit on the yield strength, in accordance with the acceptance criteria discussed in Section II.4.b. Verification is obtained that the applicant has such a control measure.

c. The information submitted by the applicant is reviewed to determine the type of insulation used and to determine its compatibility with the austenitic stainless steel used in construction of the component.

There are no compatibility concerns with the use of reflective metal insulation; the chief compatibility concern is with the use of nonmetallic insulation. A review is performed to assure that any such material specified by the applicant is in conformance with the acceptance criteria stated in Section II.4.c. Verification is obtained that the material has been chemically analyzed by methods equivalent to those prescribed in Regulatory Guide 1.36 and that evidence is obtained that the levels of leachable contaminants are such that stress corrosion of stainless steel will not result from use of the insulation.

d. The information submitted by the applicant regarding control of delta ferrite in austenitic stainless steel welds is reviewed to determine its conformance with

the acceptance criteria stated in Section II.4.d. The information submitted must state that appropriate filler metal acceptance tests have been conducted and that a certified materials test report has been received. The information should state, also, the applicant's program for compliance with the staff positions in Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

The information submitted by the applicant regarding control of electroslag weld properties for austentic stainless steel materials is reviewed for conformance with the acceptance criteria discussed in Section II.4.d.

The review of information on the control of electroslag weld properties in austenitic stainless steels is essentially the same as that discussed previously for ferritic steels. However, because electroslag-welded austenitic stainless steels have very high impact resistance and because the Code, Section III, is not concerned with impact testing of these welds, the checks are: (1) a macroetch test is used to provide assurance that the solidification pattern is in accordance with the requirement of the acceptance criteria shown in Section II.4.d, and (2) wrought stainless steel parts are solution heat treated after welding.

The review procedure for information submitted on welder qualification for limited accessibility areas, applicable to austenitic stainless steels, is the same as that for ferritic steels, which has been discussed previously under Section III.3.b.(3).

e. The procedures for review of nondestructive examination of tubular products fabricated from austenitic stainless steel are the same as those discussed for similar ferritic products in Section III.3.c of this plan, and the acceptance criteria are as shown in Section II.4.e.

5. General

If the information contained in the safety analysis reports or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional information but also the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the applicable acceptance criteria.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the review plan and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used for construction of components of the reactor coolant pressure boundary (RCPB) have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code.

"The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials, except unclad carbon and low alloy steel, is negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of the Code, Section III.

"The materials of construction for the RCPB are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

"The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," and Regulatory Guide 1.56, "Maintenance of Water Purity in BWR's," and provide reasonable assurance that the RCPB components will be adequately protected during operation from conditions that could critically lead to stress corrosion of the materials and loss of structural integrity of a component. The instrumentation and sampling provisions for monitoring reactor coolant water chemistry provide adequate measurement capability for detecting significant changes on a timely basis. Compliance with the recommendations of these Regulatory Guides constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 14 and 31.

"The fracture toughness tests required by the ASME Code, augmented by Appendix G, 10 CFR 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant pressure boundary.

"The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and NRC Regulations in establishing safe operating procedures, provides adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC Regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

"The controls imposed on welding preheat temperatures are in conformance with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." These controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment.

"The controls imposed on electroslag welding of ferritic steels are in accordance with the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties," and provide assurance that welds fabricated by the process will have high integrity and will have a sufficient degree of toughness to furnish adequate safety margins during operating, testing, maintenance, and postulated accident conditions.

"The controls imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary conform to the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.34, "Control of Electroslag Weld Properties." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with these Regulatory Guides constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 14."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants."

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. ASME Boiler and Pressure Vessel Code, Section II, Parts A, B, and C, Section III, and Section IX, American Society of Mechanical Engineers.

4. ASTM, A-262, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, American Society of Testing and Materials.

5. ASTM E 23, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, American Society of Testing and Materials.

6. ASTM E-208, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, American Society for Testing and Materials.

7. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

8. Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

9. Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

10. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants."

11. Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel."

12. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

13. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

14. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

15. Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

16. Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials."

17. Branch Technical Position MTEB 5-7 "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," appended.

18. WCAP-8577, "The Application of Preheat Temperatures After Welding Pressure Vessel Steels" Westinghouse Electric Corporation (Sept. 1975, Approved by Letter J. F. Stolz to C. Eicheldinger, June 18, 1976).

BRANCH TECHNICAL POSITION MTEB 5-7*

MATERIAL SELECTION AND PROCESSING

GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING

I. INTRODUCTION

Small, hairline cracks in austenitic stainless steel piping in boiling water reactor (BWR) facilities were observed as early as 1965. In each case, it was believed that the situation had been corrected or substantially reduced by better control of welding, contaminants and/or design modifications. In September, 1974, when the first of a series of cracks in the piping of the more modern BWRs was found at Dresden Unit No. 2., the then Atomic Energy Commission (AEC) initiated an intensive investigation to evaluate the cause, extent, and safety implications of the observed cracking. In January 1975, a special Pipe Cracking Study Group was formed to coordinate and accelerate the staff's continuing investigations of the occurrences of pipe cracking. This group included representatives of the Nuclear Regulatory Commission (NRC) and their consultants. In October 1975, the Study Group issued a report, NUREG-75/067 "Technical Report, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants." During the same general time span, the General Electric Company (GE) conducted an independent evaluation of the cracking occurrences and submitted their findings and recommendations to the NRC. This paper sets forth the NRC technical position based on the information available at this time.

Plant operating history indicates that Types 304 and 316 austenitic stainless steel piping in the reactor coolant pressure boundary of boiling water reactors are susceptible to stress corrosion cracking. Studies have shown that such cracking is caused by a combination of the presence of significant amounts of oxygen in the coolant, high stresses, and some sensitization of metal adjacent to welds. Such cracks have occurred in the heat affected zones adjacent to welds but are not expected to occur outside these areas, provided that the pipe material is properly annealed.

Pipe runs containing stagnant or low velocity fluids have been observed to be more susceptible to stress corrosion cracking than pipes containing a continuously flowing fluid during plant operation. Historically, these cracks have been identified either by volumetric examination, by leak detection systems, or by visual inspection. Because of the inherent high material toughness of austenitic stainless steel piping, stress corrosion cracking is unlikely to cause a rapidly propagating failure resulting in a loss-of-coolant accident.

Although the probability is extremely low that these stress corrosion cracks will propagate far enough to create a significant safety hazard to the public, the presence of such cracks is undesirable. Steps should therefore be taken to minimize stress corrosion cracking in BWR piping systems to eliminate this condition and to improve overall plant reliability.

*This Branch Technical Position is identical, except for typographical corrections, to the staff position given in NUREG-0313. (July 1977)

It is the purpose of this position to set forth acceptable methods to reduce the stress corrosion cracking susceptibility of BWR piping and thereby also provide an increased level of reactor coolant pressure boundary integrity. Recognizing that the most straightforward and desirable approach or methods may not be practicable, or even possible, for all plants, the bases for varying degrees of conformance to our guidelines are provided. Augmented inservice inspection and leak detection requirements are established for plants that have not fully implemented the provisions contained in Part II of this document.

II. SUMMARY OF ACCEPTABLE METHODS TO MINIMIZE CRACK SUSCEPTIBILITY

The material selection and processing guidelines listed below identify alternative acceptable methods to minimize susceptibility to stress corrosion in BWR pressure boundary piping. It is expected that adoption of these practices will result in a high degree of protection against stress corrosion cracking.

1. Corrosion Resistant Materials

   All pipe and fitting material including weld metal should be of a type and grade that has been shown to be highly resistant to oxygen-assisted stress corrosion in the as-installed condition. Unstabilized wrought austenitic stainless steel with $\geq$0.035% carbon does not meet this requirement unless all such piping including welds is in the solution annealed condition. The acceptability of alternative materials, processes, or other methods to provide an adequate degree of corrosion resistance will be made on a case-by-case basis.

2. Corrosion Resistant "Safe Ends"

   All unstabilized wrought austenitic stainless steel piping with carbon contents $\geq$0.035% should be in the solution annealed condition. If welds joining these materials are not solution annealed, they should be made between cast (or weld overlaid) austenitic stainless steel surfaces (5% minimum ferrite) or other materials having high resistance to oxygen-assisted stress corrosion. The joint design must be such that any unstabilized wrought austenitic stainless steel containing $\geq$0.035% carbon, which may become sensitized as a result of the welding process, is not exposed to the reactor coolant.

3. Other proposed methods to provide protection against stress corrosion cracking will be reviewed on a case-by-case basis.

Regulatory Guide 1.44 "Control of the Use of Sensitized Stainless Steel," dated May 1973 will be revised to provide additional guidance on acceptable practices.

III. INSERVICE INSPECTION AND LEAK DETECTION REQUIREMENTS FOR BWRs WITH VARYING CONFORMANCE TO MATERIAL SELECTION AND PROCESSING GUIDELINES

1. For plants where all ASME Code Class 1 reactor coolant pressure boundary piping subject to inservice inspections under Section XI meets the guidelines stated in Part II, no augmented inservice inspection or leak detection requirements are necessary.

2. Piping in all other plants is subject to additional inservice inspection and leak detection requirements, as described below. The degree of inspection of such piping depends on whether the specific piping runs are conforming or non-conforming, and on whether the specific piping runs are classified as "Service Sensitive." "Service Sensitive" lines are defined as those that have experienced cracking in service, or that are considered to be particularly susceptible to cracking because of high stress, or because they contain relatively stagnant, intermittent, or low flow coolant.

Examples of piping runs considered to be service sensitive include, (but are not limited to): core spray lines, recirculating by-pass lines (or "stub tubes" on plants that have removed the by-pass lines), CRD hydraulic return lines, isolation condenser lines, and shut down heat exchanger lines.

A. For non-conforming lines that are not service sensitive:

(1) Inservice inspection fo the non-conforming lines should be conducted in accordance with the schedule specified in ASME Code, Section XI - Subsection IWB, as required by the applicable examination Categories B-F and B-J, with the exception that the required examination should be completed in no more than 80 months (two thirds of the time prescribed in the schedule in the ASME Boiler and Pressure Vessel Code Section XI). If examinations conducted during the first 80 month period reveal no incidence of stress corrosion cracking, the examination schedule thereafter can revert to the schedule prescribed in Section XI of the ASME Boiler and Pressure Vessel Code.

The piping areas subject to examination, the method of examination, the allowable indication standards and examination procedures should comply with the requirements of the Edition and Addenda of the ASME Code, Section XI identified as applicable by 10 CFR Part 50, Section 50.55a, Paragraph (g), "Code and Standards."

(2) The reactor coolant leakage detection system should be operated under the following Technical Specfication requirements in order to enhance the discovery of unidentified leakage that may include through-wall cracks in austenitic stainless steel piping:

a. The source of reactor coolant leakage should be identifiable to the extent practical, using leakage detection and collection systems that meet the position described in Section C, Regulatory Position of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," or an acceptable equivalent system.

b. Plant shutdown should be initiated for inspection and corrective action when the leakage system indicates, within a period of four hours or less, an increase in the rate of unidentified leakage in excess of two gallons per minute, or when the total unidentified leakage attains a rate of five gallons per minute, whichever occurs first.

c. Unidentifed leakage should include all leakage other than:

1. Leakage into closed systems, such as pump seal or valve packing leakage that is captured, metered, and conducted to a sump or collecting tank,

2. Leakage into the containment atmosphere from sources that are specifically located and known either not to interfere with the operation of the unidentified leakage detection system, or not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

B. For non-conforming lines that are service sensitive:

(1) The leakage detection requirements described in III.A above, should be implemented.

(2) The welds and adjoining areas of bypass piping of the discharge valves in the main recirculation loops, and of the austenitic stainless steel reactor core spray piping up to and including the second isolation valve should be examined at each reactor refueling outage or at other scheduled or unscheduled plant shutdowns. Successive examinations need not be closer than six months, if shutdowns occur more frequently than six months. This requirement applies to all bypass lines whether the 4-inch valve is kept open or closed during operation.

In the event these examinations find the piping free of unacceptable indications for three successive inspections, the examination may be extended to each 36 month period (plus or minus by as much as 12 months) coinciding with a refueling outage. In these cases, the successive examination may be limited to one bypass pipe run, and one reactor core spray piping run.

(3) The welds and adjoining areas of other service sensitive piping should be examined on a sampling basis. For example, if a system consists of several branch runs with essentially symmetric piping configurations that perform similar system functions, an acceptable inspection program should include at least one, but not less than 25%, of the similar branch runs. The frequency of such examinations should be as described in 2 above. If unacceptable flaw indications are detected in any branch run, the remaining branch runs among the group should be examined.

In the event the examinations reveal no unacceptable indications within three successive inspections, the examination schedule may revert to the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components" with the exception that the required examination should be completed during each 80 month period (two-thirds the time prescribed in the schedule in the ASME Code Section XI).

(4) The method of examination, the allowable indication standards and examination procedures should comply with the requirments of the Edition and Addenda of the ASME Code, Section XI identified as applicable by 10 CFR Part 50, Section 50.55a, Paragraph (g), "Codes and Standards."

IV. IMPLEMENTATION OF MATERIAL SELECTION AND PROCESSING GUIDELINES

1. For plants that apply for a construction permit after the issue date of this document*, all ASME Code Class I reactor coolant pressure boundary lines should conform to the guidelines stated in Part II.**

2. For plants under review, but for which a construction permit has not yet been issued, all service sensitive lines should conform to the guidelines stated in Part II. Other ASME Code Class I reactor coolant pressure boundary lines should conform to Part II to the extent practicable.

3. For plants that have been issued a construction permit, ASME Code Class I reactor coolant pressure boundary lines should conform to the guidelines stated in Part II to the extent practicable.

4. For plants that have been issued an operating license, service sensitive lines should be modified to conform to the guidelines stated in Part II, to the extent practicable. Lines in which cracking is experienced should be replaced with piping that conforms to the guidelines stated in Part II.

V. GENERAL RECOMMENDATIONS

The measures outlined in Part II of this document provide for positive actions that are consistent with the current technology. The implementation of these actions should markedly reduce the susceptibility to stress corrosion cracking in BWRs. It is recognized that additional techniques are available to limit the corrosion potential of BWR coolant pressure boundary materials and improve the overall system integrity. These include plant design and operational considerations to reduce system exposure to potentially agressive environment, improve material fabrication and welding techniques and provisions for volumetric inspection capability in the design of weld joints. Specifically, consideration should be given to:

1. Minimizing the total extent of the coolant pressure boundary with special emphasis on stagnant or low flow lines.

2. Reducing the oxygen content of the primary coolant.

*Dated July 1977

**After revision, Regulatory Guide 1.44 may be used as guidance for acceptable materials, process or other methods.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 5.2.4 REACTOR COOLANT PRESSURE BOUNDARY INSERVICE INSPECTION AND TESTING

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires that components which are part of the reactor coolant pressure boundary (RCPB) shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The following areas relating to the in-service inspection program for AEC Quality Group A components of the RCPB are reviewed. These components are also ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Code Class 1 components. Inservice inspection programs are based on Section XI of the Code, "Rules for Inservice Inspection of Nuclear Power Components."

1. System Boundary Subject to Inspection

The inservice inspection (ISI) program for those portions of the reactor coolant pressure boundary consisting of Code Class 1 components is reviewed. Steam generator inservice inspection is covered separately in Standard Review Plan 5.4.2.2, "Steam Generator Inservice Inspection." Augmented inservice inspection for high energy fluid system piping between containment isolation valves is reviewed in Standard Review Plan 6.6.

2. Accessibility

The descriptive information that pertains to the general and specific provisions for access to components covered by the Code, Section XI, is reviewed. In addition, the remote access equipment needed to perform inspections in a radiation field is reviewed.

3. Examination Techniques and Procedures

The descriptive information that pertains to Section XI, Tables IWB-2500 and IWB-2600 is reviewed.

4. Inspection Intervals

The schedules of examinations and inspections in the applicant's safety analysis report (SAR) and plant Technical Specifications are reviewed. In addition, those inspections

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which are performed during the inspection interval, such as during refueling outages, are reviewed.

5. Examination Categories and Requirements

The Technical Specification tabulation of examination categories is reviewed. The SAR areas of review include category designation and the area and extent of examination of each category.

6. Evaluation of Examination Results

a. The proposed evaluation methods for any indications of structural defects detected during ISI examinations are reviewed.

b. The repair procedures proposed for components that reveal unacceptable structural defects during ISI examinations are reviewed.

7. System Leakage and Hydrostatic Pressure Tests

The descriptive information on leak tests and hydrostatic pressure tests of Code Class 1 components is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I are as follows:

1. System Boundary Subject to Inspection

The applicant's definition of the RCPB is acceptable if in agreement with the following criteria: For pressurized water reactor (PWR) and boiling water reactor (BWR) nuclear power systems, the inspection requirements of Section XI of the Code must be met for all Class 1 pressure-containing components (and their supports) except for those components excluded under IWB-1220 of Section XI. The system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:

a. The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

b. The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.

c. The reactor coolant system safety and relief valves.

2. Accessibility

The design and arrangement of system components are acceptable if in accordance with IWA-1500, "Accessibility," of Section XI. Adequate clearances for general access are demonstrated as follows:

a. Sufficient space is provided for personnel and equipment to perform inspections.

b. Provisions are made for the removal and storage of structural members, shielding components, and insulating materials, to permit access to the components being inspected.

c. Provisions are made for hoists and other handling machinery needed to handle items in b, above.

d. Provisions are made for alternative examinations in the event structural defects or indications reveal that such alternative examinations are required.

e. Provisions are made for the necessary operations associated with repair or replacement of system components and piping.

3. Examination Techniques and Procedures

The applicant's examination techniques and procedures used for ISI of the system are acceptable if in agreement with the following criteria:

a. The visual examination techniques are acceptable if in agreement with IWA-2210 of Section XI of the Code. A visual examination must be employed as a basis for a report of the general condition of the part, component, or surface. The report must include such conditions as scratches, wear, cracks, corrosion or erosion of the surfaces, misalignment or movement of the part or component, and evidence of leaking. Surface replication methods are considered acceptable provided the surface resolution is at least equivalent to that obtainable by visual observation.

b. The surface examination techniques are acceptable if in agreement with IWA-2220 of Section XI of the Code. A surface examination is required to verify the presence of surface or near surface cracks or discontinuities. The surface must not be immersed or flooded with water at the time of examination. Acceptable techniques are magnetic particle examination and liquid penetrant examination.

c. The volumetric examination methods are acceptable if in agreement with IWA-2230 of Section XI of the Code. A volumetric examination is required to indicate the presence of subsurface discontinuities with a method or technique capable of examining the entire volume of metal beneath the surface. Specific acceptable methods of volumetric examination are radiographic examination and ultrasonic examination.

d. Alternative examination methods to those given above in a, b, and c, are acceptable provided the results are equivalent or superior. The acceptance standards for these alternative methods are given in Section XI, IWB-3100, "Evaluation of Nondestructive Examination Results."

4. Inspection Intervals

The inservice inspection program is acceptable if the required examinations and pressure tests are completed during each ten-year interval of service, hereafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of IWA-2400, "Inspection Intervals," of Section XI of the Code.

It is intended that inservice examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval. Except as specified in Table IWB-2500, "Examination Categories," at least 25% of the required examinations must be completed by the expiration of one-third of the inspection interval. Credit is allowed for no more than 33-1/3% of the required ISI even though additional examinations are completed during this period. At least 50% of the required examination must be completed by the expiration of the second one-third of the inspection interval. Credit is allowed for no more than 66-2/3% of the required inspections. The remaining required examinations shall be completed by the end of the inspection interval.

5. Examination Categories and Requirements

The examination categories and requirements as specified in the SAR are acceptable if in agreement with the criteria of IWB-2500 and IWB-2600 of Section XI of the Code. Every area subject to examination falls within one or more of the examination categories indicated in Table IWB-2500 and must be examined at least to the extent specified. The method of examination for the components and parts of the pressure-containing and pressure-retaining boundaries that are listed in the requirements of IWB-2600 of Section XI are tabulated in Table IWB-2600.

6. Evaluation of Examination Results

a. The standards for examination evaluation are acceptable if in agreement with the requirements of Section XI, IWB-3000, "Standards for Examination Evaluations." The applicant's program for flaw evaluation is acceptable if it agrees with Table IWB-3410, "Evaluation Standards."

b. The proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if in agreement with the requirements of Section XI, IWB-4000, "Repair Procedures." The criteria that establish the need for repair or replacement are described in Section XI, IWB-3000.

7. System Leakage and Hydrostatic Pressure Tests

The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program agrees with the requirements of Section XI, IWB-5000, "System Leakage and Hydrostatic Pressure Tests." IWB-5222, "System Hydrostatic Test Pressure," presents criteria and a table of equivalent test temperatures versus test pressures at which the system must be tested. The applicant's program is acceptable if in agreement with IWB-5222 in regard to the temperature-pressure relationship of the system at test, and if in agreement with the Technical Specification requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. In some cases, these limitations may be more severe than IWB-5222.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. System Boundary Subject to Inspection

The information furnished in the SAR is reviewed for agreement with Section II.1 of this plan, and to verify that any differences between the applicant's definition of the RCPB and Section II.1 are identified and justified by the applicant, e.g., "Pressurizer: not applicable, as plant is a BWR." or, "no longitudinal welds in beltline region as vessel is constructed of forged rings."

2. Accessibility

The descriptive information concerning accessibility furnished in the SAR is reviewed for compliance with Section II.2. The reviewer verifies that the clearances supplied

for general access to the system component listed in Table IWB-2500 of Section XI are adequate.

The reviewer verifies that adequate provisions are made for remote inspection of those components affected by radiation fields after plant start-up. These components include the beltline welds and reactor vessel nozzle interior surfaces. The reviewer verifies that remote inspection devices proposed for periodic inservice inspections will be used for the preservice baseline inspection program to demonstrate feasibility.

3. Examination Techniques and Procedures

The reviewer verifies that the examination techniques described by the applicant are the same as those in Section II.3. If alternative examination methods are proposed by the applicant, they are reviewed to verify that the results are equivalent or superior to those in IWA-2210, 2220, and 2230 of Section XI, and that the acceptance standards of IWB-3100 of Section XI are met.

4. Inspection Intervals

The Technical Specification program for inservice inspection is reviewed to establish that the inspection schedule for every area and component in the program is in agreement with Section II.4.

5. Examination Categories and Requirements

The descriptive information in the SAR and the Technical Specification ISI program are reviewed to establish that the applicant followed the requirements of Section II.5. The reviewer determines that the table supplied in the Technical Specifications follows Table IWB-2600 of Section XI where it is applicable to the given reactor system, and that the table contains the following headings and applicable information: Examination Category, Components and Parts to be Examined, Method, Extent and Frequency of Examinations, and Comments.

6. Evaluation of Examination Results

The criteria statements provided by the applicant are reviewed for agreement with Section II.6 as follows:

a. The reviewer verifies that the applicant's criteria incorporate IWB-3000 of Section XI regarding standards for examination evaluation.

b. The reviewer verifies that the applicant's criteria incorporate IWB-4000 of Section XI regarding repair procedures.

7. System Leakage and Hydrostatic Pressure Tests

The reviewer determines that the Technical Specification on hydrostatic pressure testing for system leakage of the RCPB adheres to Section II.7 of this plan and incorporates the table in IWB-522, Section XI. The Technical Specification on operating limitations during heatup, cooldown, and system hydrostatic pressure testing must be referenced.

IV. EVALUATION FINDINGS

The reviewer verifies that adequate information is provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. The applicant has stated that the designs of Code Class 1 components of the reactor coolant pressure boundary incorporate provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that methods will be developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel. The conduct of periodic inspections and leakage and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Code provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of General Design Criterion 32."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."
2. ASME Boiler and Pressure Vessel Code, Section XI, Section III, "Nuclear Power Plant Components," and "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

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Section 5.2.5 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

General Design Criterion 30 (Ref. 1) requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

The areas of the SAR relating to the reactor coolant pressure boundary (RCPB) leakage detection systems are reviewed. The descriptive information and supporting figures, tables, and graphs are reviewed to establish that sufficient information is provided to permit a reasonable evaluation of the applicant's compliance with Regulatory Guide 1.45 (Ref. 2), as follows:

1. Collection of Identified Leakage

A limited amount of leakage is expected from components of the RCPB within the containment, such as valve stem packing glands, circulating pump shaft seals, and other equipment that cannot practically be made completely leaktight. The reactor vessel closure seals and safety and relief valves should not leak significantly; however, leakage occurring via these paths or via pump and valve seals is detectable and collectable and, to the extent practical, should be isolated from the containment atmosphere so as not to mask any potentially serious leak should it occur. These leaks are known as "identified leakage" and are piped to tanks or sumps so that the flow rate can be established and monitored during plant operation. The provisions for collecting and monitoring leakage from known leak sources are reviewed.

2. Unidentified Leakage to Containment

Uncollected leakage to the containment atmosphere increases the humidity of the containment. The moisture removed from the atmosphere by air coolers together with any associated liquid leakage to the containment is known as "unidentified leakage" and is collected in tanks or sumps where the flow rate can be established and monitored during plant operation. Unidentified leakage to the containment atmosphere should be kept to a minimum to permit the leakage detection systems to detect positively and rapidly a small increase in flow rate. Identified and unidentified

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leakages should be separated so that a small unidentified leakage that is of concern will not be masked by a larger acceptable identified leakage. Provisions for the detection and control of leakage to containment are reviewed.

3. Leakage Detection Methods

Continuous monitoring of both identified and unidentified leakage rates is important. Effective systems for detecting and locating unidentified leakage are needed. The following describes some detection methods commonly used.

The primary monitors determine flow rates and flow rate changes to tanks and sumps. Methods to indicate when and where coolant is being released to the containment atmosphere include detection of changes in airborne particulate radioactivity, airborne gaseous radioactivity, containment atmosphere humidity, pressure, and temperature, condensate flow rate from air coolers.

4. Intersystem Leakage

Substantial intersystem leakage from the RCPB to other systems across passive barriers or valves is not expected. However, should such leakage occur, it should be detectable by the alarm and detection methods which are employed. For example, steam generators in pressurized water reactors (PWR's) are monitored to detect tube sheet leaks.

Since intersystem leakage does not release reactor coolant to the containment atmosphere, detection methods include monitoring of radioactivity in the connected systems where the flow is through the containment boundary, and monitoring of airborne radioactivity where such systems are vented outside the containment boundary. Another important method of obtaining indications of uncontrolled or undesirable intersystem flow is the use of a water inventory balance, designed to provide appropriate information such as abnormal water levels in tanks and abnormal water flow rates.

5. System Sensitivity and Response Time

Since leakage detection methods or systems differ in sensitivity and response time, prudent selection of detection methods should include a sufficient number of systems to ensure effective monitoring during periods when some detection systems may be ineffective or inoperable. Some of these systems should serve as early alarm systems which signal the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required. It is essential that leakage detection systems have the capability to detect significant RCPB leakage as soon after occurrence as practical to minimize the potential for a gross boundary failure. Cracks that might develop and penetrate the RCPB wall are expected to exhibit very slow growth, and to afford ample time for a safe and orderly plant shutdown after a leak is detected. An early warning signal is necessary to permit proper evaluation of all unidentified leakage.

6. Seismic Capability of Systems

Since nuclear power plants may be operating at the time an earthquake occurs and may continue to operate after earthquakes, the leakage detection systems should be designed to continue functioning after seismic events. If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, it is important that the operator be able to assess the condition within the containment quickly. The proper functioning of at least one leakage detection system is essential in evaluating the seriousness of the condition within the containment in the event leakage has developed in the RCPB. The MEB reviews the seismic qualification of the electrical and instrumentation portion of the leak detection system in SRP 3.10 (Ref. 3).

7. Quantitative Interpretation of Indicators and Alarms

It is important to be able to associate a signal or indication of a departure from the normal operating conditions with a quantitative leakage flow rate. Except for flow rate or level change measurements from tanks, sumps, or pumps, signals from other leakage detection systems do not provide information readily convertible to a common denominator. Approximate relationships converting these signals to units of water flow are formulated to assist the operator in interpreting signals. The instrumentation associated with the leak detection system is reviewed by EICSB in SRP 7.5 (Ref. 4). Procedures for operator evaluation of leakage conditions are reviewed by RSB.

8. Testability

Provisions for testing the various systems during plant operation should be provided. EICSB ensures that leakage detection equipment is tested and calibrated in compliance with IEEE Std. 279-1971 (Ref. 5).

9. Technical Specifications

RSB reviews the limiting conditions for operation that appear in the technical specifications. Leakage limits for unidentified and total leakage, maximum time allowed to operate after a leak is discovered, and action to take in the event of instrument malfunction are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are stated in Regulatory Guide 1.45 (Ref. 2). According to this guide the source of reactor coolant leakage should be identifiable to the extent practical. Reactor coolant pressure boundary leakage detection and collection systems are acceptable if they are in accordance with the following:

1. Collection of Identified Leakage

Leakage to the primary reactor containment from identified sources should be collected or otherwise isolated so that the flow rates are monitored separately from unidentified leakage, and the total flow rate can be established and monitored.

2. Collection and Monitoring of Unidentified Leakage

Leakage to the primary reactor containment from unidentified sources should be collected and the flow rate monitored with an accuracy of one gallon per minute (gpm) or better.

3. Leakage Detection Methods

At least three separate detection methods should be employed and two of these methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be selected from either the monitoring of condensate flow from air coolers, or monitoring of airborne gaseous radioactivity.

Humidity, temperature, or pressure monitoring of the containment atmosphere are to be considered as alarms or indirect indications of leakage to the containment.

4. Intersystem Leakage

Provisions should be made to monitor systems connected to the RCPB for signs of intersystem leakage. Detection methods include radioactivity monitoring and indicators to show abnormal water levels or flow in the potentially affected systems and unaccountable increases in reactor coolant make-up flow.

5. System Sensitivity and Response Time

The sensitivity and response time of each leakage detection system employed for monitoring unidentified leakage to the containment should be adequate to detect an increase in leakage rate, or its equivalent, of one gpm in less than one hour.

6. Seismic Capability of Systems

The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown and the airborne particulate radioactivity monitoring system should be capable of remaining functional when subjected to the safe shutdown earthquake (SSE).

7. Indicators and Alarms

Indicators and alarms for each leakage detection system should be provided in the main control room and procedures for converting various indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for the independent variables such as, in the case of an air particulate monitor, the isotope being monitored, plateout, and decay rate. Each system should be set to alarm on an increase in leakage of 1 gpm above the background level determined at the time of calibration.

8. Testing

The leakage detection systems should be equipped with provisions to permit calibration and operability tests during plant operation.

9. Technical Specifications

The technical specifications should include, in the limiting conditions for operation, the maximum permissible total and unidentified leakage, and address the availability of the leakage detection systems to ensure adequate coverage at all times. The leakage limits are established on the basis of current practice for similar types of nuclear steam supply systems.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For the operating license (OL) review, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and allowable leakage rates.

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. Collection of Identified Leakage

   Information concerning the collection of identified leakage is reviewed for agreement with Acceptance Criterion II.1. The reviewer verifies that the SAR description of the reactor vessel flange leakage monitoring, leakage monitors for other vessel flanges, and valve and pump seal leakage monitors is complete, and that this monitored leakage will be collected in tanks or sumps where its rate of accumulation will be summed to obtain an identified leak rate. The reviewer should establish that the identified leakage is not only collected and monitored but in such a fashion as to prevent identified leakages from masking unidentified leaks.

2. Collection and Monitoring of Unidentified Leakage

   Information concerning the collection and monitoring of unidentified leakage is reviewed for agreement with Acceptance Criterion II.2.

3. Leakage Detection Methods

   The information describing the number of systems and operating principles of each system, including schematic diagrams, is reviewed to assure that sufficient information is provided to comply with Acceptance Criterion II.3. The review consists of a side-by-side comparison of the applicant's methods and the acceptance criterion and a determination that the number and type of methods provided are acceptable.

4. Intersystem Leakage

   Information describing the intersystem leakage detection system is reviewed for compliance with Acceptance Criterion II.4. The reviewer determines that radiation monitoring systems have been provided for possible intersystem leakage paths, including all auxiliary cooling systems interfacing with the primary coolant, such as heat exchangers or steam generators in PWR plants. The reviewer determines that intersystem leakage monitors are sensitive to the radiation emitted by fission products such as I-131 and the radioactive isotopes of xenon and krypton. The reviewer assures that the monitoring systems and procedures to detect and control leakage are capable of keeping intersystem leakage within the limits assumed in accident analyses. For example, steam generator tube and tube sheet leaks should be detected and corrective action taken before the contamination of the secondary coolant exceeds that assumed in evaluating the steam generator tube accident without offsite power (see SRP 15.6.3).

5. System Sensitivity and Response Time

The reviewer determines that all components for the detection of unidentified leakage called for by Acceptance Criterion II.3 meet the sensitivity and response time of Acceptance Criterion II.5. Currently used systems that have been found acceptable measure leakage either directly in gpm, such as sump monitors and containment air cooler condensate monitors or indirectly in units of radiation, μCi/cc, in the containment atmosphere.

The two most-used radiation sensitive monitors are the air particulate monitor (APM) and the radiogas monitor (RGM). The threshhold sensitivity of the APM is 10^{-9} μCi/cc of containment volume and the RGM can sense 10^{-6} μCi/cc. The APM is 1000 times more sensitive than the RGM, hence its selection in Regulatory Guide 1.45 as one of the two systems a plant should have without any alternate choice. Background activity levels corresponding to assumed normal conditions of primary coolant leakage and failed fuel fraction may be used to evaluate the compliance to Acceptance Criterion II.5.

6. Seismic Capability of Systems

The SAR should state that the leakage detection systems meet the seismic capability recommendations of Acceptance Criterion II.6. The reviewer verifies that the leakage detection systems will remain functional for all seismic events not requiring a shutdown. In addition, the reviewer verifies that the APM can function after the safe shutdown earthquake. The reviewer determines that the applicant has provided the capability to take grab samples of the containment atmosphere on a periodic basis and manually analyze these samples in his radiochemistry laboratory for particulate activity and to correlate the data to primary system leakage.

7. Indicators and Alarms

Information concerning the indicators and alarms is reviewed for compliance with Acceptance Criterion II.7. The reviewer verifies that all of the leakage detection systems have readouts in the control room and are provided with alarms. Direct reading systems, such as sumps, will normally indicate in gpm. The indirect reading systems, such as the APM, will indicate in counts per minute. The reviewer determines that control room operators will have a chart or graph that permits rapid conversion of count rate into gpm, that the conversion procedures take into account the isotope being monitored and the activity of the primary coolant, and that the plant will maintain a running record of background leakage, so that its effect may be subtracted from any sudden increases in leak indication, which may be "unidentified" leakage and require prompt action. If monitoring is computerized, backup procedures should be available to the operator.

8. Testing

Information concerning operability testing and system calibration during plant operation is reviewed for compliance with Acceptance Criterion II.8. The reviewer determines that the radiation monitoring systems have a radioactive source built into the system (the SAR refers to this feature as a "check source") to permit system test and calibration during operation. He also determines that the flow of "identified" leakage, which may amount to as little as 0.05 gpm or as much as 0.25 gpm, representing a total

daily flow of between 72 and 360 gallons, will be used to provide an operability check during operation for the sump monitoring systems and the containment air cooler condensate flow monitors. The directly measured quantity of flow thus obtained from the sump and air cooler monitors can be used to calibrate the radiation monitoring systems.

9. Technical Specifications

The technical specifications are reviewed for compliance with Acceptance Criterion II.9. The reviewer compares the proposed technical specification limits for unidentified and total allowable leakage to the design basis as determined in the review of items 1-5 and 7. In addition, the reviewer determines that the availability of various components of the leakage detection system and the action to be taken if a component becomes inoperative are addressed in the technical specifications. The availability of the leakage detection components has to be reviewed on a case-by-case basis due to the large number of possible component combinations, multiplicity of systems, use or lack of redundancy, or the ability to use systems not specifically called out as "leakage detection systems" but still able to perform this role as a secondary function to the primary design use of such system. An example would be containment vent radiation monitors. A suggested technical specification regarding availability is as follows:

> "Both the sump monitoring and air particulate monitoring systems shall be operable during reactor power operation. If either system becomes inoperable for any reason, reactor power operation is permissible only during the succeeding seven days unless the system is made operable sooner. If the above conditions cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours."

10. General

Should the leakage detection system submittal contain additional data and analyses as a basis to support a system not discussed in Regulatory Guide 1.45, the reviewer should evaluate the applicant's data and analyses to determine if the proposed system has leakage detection capabilities comparable to those of the standard systems. The reviewer can also find guidance in review procedures III.1 through 9, above, and in other SAR's where applicants may have proposed similar alternate systems.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided in accordance with the requirements of this review plan, and that his evaluation provides the basis for conclusions of the following type, which should be included in the staff's safety evaluation report:

> "Coolant leakage within the primary containment may be an indication of a small through-wall flaw developed in the reactor coolant pressure boundary (RCPB).
>
> "The leakage detection system provided will include sufficiently diverse leak detection methods, with adequate sensitivity to measure small leaks and to identify the leakage sources within practical limits, with the aid of suitable control room alarms and read-outs. The major systems are the containment atmosphere particulate and radiogas monitors, and level indicators on the containment sumps. Indirect indications of leakage are obtainable from the containment pressure, humidity, and temperature indicators.

"The leakage detection systems provided to detect leakage from components of the reactor coolant pressure boundary furnish reasonable assurance that structural degradation, which may develop in pressure-retaining components of the RCPB and result in coolant leakage during service, will be detected on a timely basis, so that corrective actions can be made before such degradation could become sufficiently severe to jeopardize the safety of the system, or before the leakage could increase to a level beyond the capability of the makeup systems to replenish the coolant loss. The systems are in compliance with the recommendations of Regulatory Guide 1.45 and satisfy the requirements of General Design Criterion 30, Appendix A of 10 CFR Part 50."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, Criterion No. 30, "Quality of Reactor Coolant Pressure Boundary."

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3. Standard Review Plan 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment."

4. Standard Review Plan 7.5, "Safety-Related Display Instrumentation."

5. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.3.1 REACTOR VESSEL MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires that the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

The following areas relating to reactor vessel materials are reviewed:

1. Materials Specifications

The material specifications used for the reactor vessel and applicable appurtenances such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings are reviewed and their adequacy for use in the construction of such components is assessed on the basis of the material, mechanical, and physical properties, the effects of irradiation on these materials, their corosion resistance, and fabricability. Similarly, the specifications for austenitic steel and nonferrous metals specified for the above applications are reviewed with respect to mechanical properties, stress-corrosion resistance, and fabricability.

2. Special Processes Used for Manufacture and Fabrication of Components

Information submitted by the applicant for any special process used in the manufacture of the product forms supplied, and their fabrication into the reactor vessel or any of its appurtenances is reviewed, and the capability of these processes to provide components with suitable mechanical and physical properties is assessed. The effects of such special processes on the stress-corrosion characteristics of the material, and any aspect of the process which could cause special requirements for nondestructive examinations are reviewed.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

3. Special Methods for Nondestructive Examination

Nondestructive examination methods differing from those described in the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III are reviewed. Attention is directed towards calibration methods, instrumentation, methods of application, sensitivity, reliability, and standards used.

4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels

Information on special controls and special processes for welding ferritic steels and austenitic stainless steels is reviewed, and their adequacy is assessed. The extent to which the controls and processes deviate from code rules is reviewed. Information on welding of safe-ends during the fabrication of dissimilar metal joints is given particular attention and details of the methods, processes, and materials used are reviewed.

5. Fracture Toughness

Fracture toughness of ferritic materials used for reactor vessels and appurtenances thereto is reviewed to assure that such components will behave in a nonbrittle manner and that the probability of rapidly propagating fracture will be minimized under operating, maintenance, testing, and postulated accident conditions. The review includes the descriptions of the fracture toughness tests performed on all ferritic materials used for the reactor vessel and appurtenances thereto, and includes transverse Charpy V-notch impact test specimens, dropweight test specimens, and any other test specimens included by the applicant.

The test procedures specified by the applicant are reviewed and their adequacy is confirmed.

The composition of ferritic materials employed for the reactor vessel is reviewed and the amount of residual elements such as copper, sulfur, and phosphorous is checked. The results of impact tests performed on base material, weld metal, and heat-affected zones are reviewed, and the scope of the testing is checked, particularly in the area of the reactor vessel beltline region where radiation effects on the material are most significant.

Fracture toughness of the materials employed is characterized by its reference temperature, RT_{NDT}. This temperature is the higher value of the nil-ductility temperature (NDT) from the dropweight test, or the temperature that is $60^{o}F$ below the temperature at which Charpy V-notch impact test data meet a specified toughness level. The information submitted is checked to ensure that the RT_{NDT} of the materials is included with the data and test results for impact testing.

6. Materials Surveillance

Reactor vessel material surveillance must be performed to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline

region of water-cooled power reactors, resulting from exposure to neutron irradiation and the thermal environment. Under the surveillance programs, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

7. Reactor Vessel Fasteners

The materials for the stud bolts, washers, and nuts, or other fasteners used to hold the reactor vessel head are reviewed to determine their adequacy. Mechanical properties, including fracture toughness, are checked to ensure that all requirements are met. Lubricants or surface treatments used are reviewed to assure that the studs will be resistant to stress-corrosion cracking under the environmental conditions during service and shutdowns. The adequacy of the destructive testing procedures used to ensure initial integrity is reviewed, along with the applicable acceptance criteria.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Material Specifications

Acceptable material specifications are those listed in the Code, Section III, Appendix I, and are presented in detail in Section II, Parts A, B, and C.

The acceptability of materials not specified in the Code are considered on an individual basis. Their suitability is evaluated on the basis of data submitted in accordance with the requirements of Code Section III, Appendix IV-1400. These data must include information on mechanical properties, weldability, and physical changes of the material.

2. Special Processes Used for Manufacturing and Fabrication

The reactor vessel and its appurtenances are fabricated and installed in accordance with Code Section III, Paragraph NB-4100. The manufacturer or installer of such components is required to certify, by application of the appropriate Code Symbol and completion of an appropriate data report in accordance with Code Section III, Paragraph NA-8000, that the materials used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000.

3. Special Methods for Nondestructive Examination

The acceptance criteria for examination of the reactor vessel and its appurtenances by nondestructive examination are those specified in Code Section III, NB-5000, for normal methods of examination. When special techniques or procedures are developed, they must be equivalent or superior to the techniques described in Appendix IX-3000 of Code Section III, and must be proven so, by demonstration on the specific type of component part.

4. Special Controls and Processes for Welding Ferritic Steel Components and Austenitic Stainless Steel Components

Only those welding processes capable of producing welds in accordance with the welding procedure qualification requirements of Code Sections III and IX may be used. Any process used shall be such that the records required by NB-4320 of Section III can be made, with the exception of stud welding which is acceptable only for minor nonpressure attachments.

These requirements supplement those shown in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials," concerning welding controls for ferritic and austenitic stainless steels.

5. Fracture Toughness

The acceptance criteria for this area of review are based on General Design Criterion 31, Code Section III, and Appendix G of 10 CFR Part 50.

The reactor vessel and appurtenances thereto which are made of ferritic materials must meet the following requirements for fracture toughness during system hydrostatic tests, during conditions of normal operation, and during anticipated operational occurrences:

a. The materials shall meet the acceptance standards of Paragraph NB-2330 of the Code which states that at a temperature not greater than (T_{NDT} + 60°F) each Charpy C_v specimen tested (per NB-2321.2) shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs of absorbed energy.

b. When these requirements are met, T_{NDT} is defined as the reference temperature, RT_{NDT}. In the event that the above requirements are not met, additional C_v notch impact tests are performed (in groups of three specimens, per NB-2321.2) to determine the temperature T_{CV} at which they are met. In this case the reference temperature $RT_{NDT} = T_{CV} - 60°F$. Thus the reference temperature RT_{NDT}, is the higher of T_{NDT} and (T_{CV} - 60°F).

c. When a C_v impact test has not been performed at (T_{NDT} + 60°F), or when the C_v impact test at (T_{NDT} + 60°F) does not exhibit a minimum of 50 ft-lbs and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lbs and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v impact tests performed. In addition to the above criteria, the requirements of Section IV, A.2, 3, and 4 and IV.B of Appendix G of 10 CFR Part 50 must be met.

Standard Review Plan 5.3.2, "Pressure-Temperature Limits," discusses the requirements of Section IV. A.2, 3, and 4 of Appendix G of 10 CFR Part 50 in detail.

The acceptance criteria discussed in Section IV.B of Appendix G of 10 CFR Part 50 state that reactor vessel beltline materials shall have a minimum upper-shelf energy as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with Paragraphs NB-2322.2(4) and NB-2322.6 of the Code, Section III, of 75 ft-lbs unless it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper-shelf fracture energy are adequate.

6. Materials Surveillance

The material surveillance criteria are given in Appendix H to 10 CFR Part 50, Section II.

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that have been verified by experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E > 1 Mev) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

Reactor vessels constructed of ferritic materials which do not meet these conditions shall have their beltline regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," ASTM Designation: E-185-73, except as modified by Appendix H to 10 CFR Part 50.

The surveillance program shall meet the following requirements:

a. Surveillance specimens must be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G of 10 CFR Part 50. The speciment types must comply with the requirements of Section III.A of Appendix G, except that dropweight specimens are not required.

b. Surveillance capsules containing the surveillance specimens must be located near but not attached to the inside vessel wall in the beltline region, so that the neutron flux received by the specimens is at least as high but not more than three times as high as that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. The design and location of the capsules must permit insertion of replacement capsules. Accelerated irradiation capsules, for which the calculated neutron flux will exceed three times the calculated maximum neutron flux at the inside wall of the vessel, may be used in addition to the required number of surveillance capsules specified in Section II.C.3 of Appendix H, 10 CFR Part 50.

c. The required number of capsules, which will vary from three to five depending upon the adjusted reference temperature at the end of the service lifetime of the reactor vessel, and their withdrawal schedules, must be in accordance with the requirements of Section II.C.3 of Appendix H of 10 CFR Part 50.

d. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

7. Reactor Vessel Fasteners

The Code, Section III, Appendix G to 10 CFR Part 50, and Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," define the acceptance criteria for reactor vessel studs and fasteners.

Materials for reactor vessel studs (and other fasteners) that are considered suitable are SA-540 Grades B-23 and B-24, SA-193 Grade B-7, SA-194 Grade 7, and SA-320 Grade L-43.

The fastener material should not have an ultimate tensile strength over 170 ksi, and the fracture toughness tests and acceptance levels of Paragraph IV.A.4 of Appendix G to 10 CFR Part 50 must be met.

Surface treatments, plating, or thread lubricants used must be shown to be compatible with the materials, and stable at operating temperatures.

Nondestructive examination should be performed according to Section III of the Code, Subarticle NB-2580, including the additional recommendations given in Regulatory Guide 1.65.

These are:

a. The stud bolts and nuts should be ultrasonically examined after final heat treatment and prior to threading.
b. The ultrasonic examination (paragraph NB-2584) should be conducted according to ASME Specification SA-388, "Ultrasonic Examination of Heavy Steel Forgings."
c. The calibration standard used to establish the first back reflection for the ultrasonic testing should be based on good sound representative material. To assure that the material is representative, the selection of the standard should be based on a preliminary ultrasonic examination of a number of specimens (a minimum of three per standard).
d. The magnetic particle or liquid penetrant examination (paragraph NB-2583) should be performed on the studs and nuts after final heat treatment and threading.
e. The requirements of paragraph NB-2585 should be applied to all closure stud bolts and nuts.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in Section I of this plan, the review procedure is as follows:

1. Materials Specifications

The material specifications for the reactor vessel and its appurtenances are compared with the acceptable specifications listed in the Code, Section III, Appendix I, and Section II, Parts A, B, and C.

Any materials not listed in the Code, or any deviations in a listed specification, are clearly identified, and the bases for deviation or nonconformance evaluated. A study of the suitability of the material and comparisons with precedents set in earlier cases enable the reviewer to determine the acceptability of the proposed exceptions. In those instances where the Materials Engineering Branch has taken exception to the use of a specific material, or questions certain aspects of a specification, the applicant is advised which material is not acceptable, and the reason for disapproval.

2. Special Processes Used for Manufacture and Fabrication of Components

Information on special processes used for manufacture and fabrication of the reactor vessel and its appurtenances are reviewed to (1) identify each special process, (2) determine whether there are any code restrictions on its use, (3) establish the adequacy of the process in providing components with suitable mechanical and physical properties, (4) establish the effects of such processes on the stress-corrosion characteristics of the material, and (5) identify whether special requirements for nondestructive examination are needed if the process is used.

Since there are no specific code requirements on the use of special processes, the suitability of a process is assessed on the basis of service experience with similar parts fabricated by the process being reviewed.

3. Special Methods Used for Nondestructive Examination

Section V of the Code includes methods for performing nondestructive examinations to detect surface and internal discontinuities when these methods are referenced by Section III of the Code. They include the following methods: radiographic, magnetic particle, liquid penetrant, and ultrasonic. The methods as described are applicable to most geometric configurations and materials encountered in fabrication, and are applied for normal conditions. However, special configurations and materials may require modified methods and techniques. If such special procedures are developed, the reviewer must determine that they are equivalent or superior to the techniques described in Section V of the Code, and are capable of producing meaningful results under the special conditions.

Such special procedures may be modifications or combinations of methods described in Section V, or may be entirely different, but the reviewer verifies that they have been proven by demonstration to result in an examination capable of detecting discontinuities under the special conditions to the same extent that applicable normal techniques which are included in Section V would result in detection of discontinuities under normal conditions.

Such special procedures are submitted to the authorized inspector or inspecting agency for review and approval prior to use.

4. Special Controls and Processes for Ferritic Steels and Austenitic Stainless Steel

The controls on welding of ferritic steels and austenitic stainless steels discussed in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials," are considered applicable to welding of the reactor vessel and its components. The reviewer verifies that any special welding control or special welding process is able to conform to the qualification requirements of the Code, Section IX, or that justification is made for this deviation.

In the event that this information is lacking in the SAR, the reviewer prepares a request for additional information, for transmittal to Reactor Projects. Such a request not only identifies the additional information required, but also lists the

changes needed in the SAR. Subsequent amendments received in response to these requests are reviewed for compliance with the stated criteria.

5. Fracture Toughness

The information submitted by the applicant relative to tests for fracture thoughness is reviewed for conformance with the Code, Section III, Paragraph NB-2320, and Appendix G of 10 CFR Part 50.

These tests include Charpy V-notch impact tests and dropweight tests. A description of the tests is reviewed, and the location of the test specimens and their orientation are verified.

Information regarding calibration of instruments and equipment are reviewed for conformance to Code Section III, Paragraph NB-2360.

In the event that none of the fracture toughness tests have been performed, the preliminary safety analysis report (PSAR) must contain a statement of the applicant's intention to perform this work in accordance with Code Section III, NB-2300 and Appendix G of 10 CFR Part 50.

The final safety analysis report (FSAR) is reviewed to assure that all the impact tests shown in NB-2320 have been performed. The results of the tests shall be in accordance with the acceptance criteria shown in II.5 of this plan.

If the information contained in the SAR does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, the reviewer prepares a request for additional information for transmittal to Reactor Projects. Such requests not only identify the additional information required, but also specify the changes needed in the SAR or the plant Technical Specifications to meet acceptance criteria. Subsequent amendments received in response to these requests are reviewed for compliance with the stated criteria.

6. Materials Surveillance

The reviewer verifies that the information contained in the SAR and the Technical Specifications is complete enough to determine that the surveillance program will comply with Appendix H, 10 CFR Part 50. The following information must be provided as a minimum.

a. The reviewer verifies that the PSAR states the end of life fluence calculated for the vessel beltline, the maximum predicted shift in reference transition temperature (RT_{NDT}), the number of capsules, and the number and types of specimens to be placed in the capsules, and that the program is in compliance with ASTM E 185-73 and Appendix H, 10 CFR Part 50.

b. The reviewer verifies that the FSAR provides the information listed above, and in addition, includes results of fracture toughness tests and chemical analyses of all materials in the beltline region, and provides the information needed by the reviewer to evaluate the adequacy of the program.

7. Reactor Vessel Fasteners

The reviewer verifies that the information in the SAR covers all requirements for reactor vessel studs and other fasteners, as described in the previous section. For FSAR's, the results of tensile and fracture toughness tests performed on the fastener materials are checked to ensure that all requirements are met.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the review plan, and that his evaluation supports the conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Special requirements of the applicant with regard to control of residual elements in ferritic materials have been identified and are considered acceptable.

"Special processes used for manufacture or fabrication of the reactor vessel and its appurtenances have been identified, and appropriate data reports on each process as required by Section III of the ASME Code have been submitted by the applicant. Since certification has been made by the applicant that the materials and fabrication requirements of Section III of the Code have been complied with, the special processes used are considered acceptable.

"Special methods used for nondestructive examination of the reactor vessel and its appurtenances have been identified, and have been found equivalent or superior to the techniques described in Appendix X of Code Section III. Demonstrations have been made using these special techniques, and have satisfied all requirements of the Code. The special methods of nondestructive examination are deemed acceptable.

"Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of Code Sections III and IX. The controls imposed on welding preheat temperatures are in conformance with the requirements of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Allow Steels," and provide reasonable reassurance that cracking of components made from low alloy steels will not occur during fabrication, and will minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment. The controls imposed on electroslag welding of ferritic steels are in accordance with the requirements of Regulatory Guide No. 1.34, "Control of Electroslag Weld Properties," and provides assurance that welds fabricated by the process will have high integrity, and will have a sufficient degree of toughness to furnish adequate safety margins during operating, testing, maintenance, and postulated accident conditions. The controls imposed upon austenitic stainless steel welds are in conformance with Regulatory Guide No. 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

"The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G of the Code as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the Code and AEC Regulations, will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and AEC Regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

"Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation have been assessed properly, and adequate safety margins against the possibility of vessel failure are provided as the material surveillance requirements of ASTM E 185-73 and Appendix H, 10 CFR Part 50 are met. Compliance with these documents assures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of General Design Criterion 31."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants."

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4. ASME Boiler and Pressure Vessel Code, Section II, Section III, Section V, and Section IX, American Society of Mechanical Engineers.

5. ASTM E-23-72, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

6. ASTM E-185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials.

7. ASTM E-208-69, "Standard Method for Conducting Drop-weight Test to Determine Nil-ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

8. Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

9. Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

10. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

11. Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

12. Regulatory Guide 1.31, "Control of Stainless Steel Welding."

13. Branch Technical Position MTEB 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," appended to Standard Review Plan 5.2.3.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.3.2 PRESSURE-TEMPERATURE LIMITS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

1. Pressure-Temperature Limits

The following pressure-temperature limits imposed on the reactor coolant pressure boundary (RCPB) during operation and tests are reviewed to assure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components of the RCPB, as required by General Design Criterion No. 31:

a. Pressure-temperature limits for preservice hydrostatic tests.
b. Pressure-temperature limits for inservice leak and hydrostatic tests.
c. Pressure-temperature limits for heatup and cooldown operations.
d. Pressure-temperature limits for core operation.

II. ACCEPTANCE CRITERIA

1. Applicable Regulations, Codes, and Basis Documents

Appendices G and H of 10 CFR Part 50 describe the conditions that require pressure-temperature limits and provide the general basis for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Appendix G, "Protection Against Non-ductile Failure," during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins whenever the reactor core is critical (except for low-level physics tests).

The Code, Section III, specifies fracture toughness testing requirements for ferritic materials. Appendix G of the Code provides a basis for determining allowable pressure-temperature relationships for normal, upset, and test conditions.

Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials," provides the detailed technical basis for the code requirements.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

2. Technical Bases

a. The principles of linear elastic fracture mechanics (LEFM) are used to determine safe operational conditions. The basic parameter of LEFM is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. An analytical method is used to determine the effects of real or postulated flaws. The minimum K_I that can cause failure is defined as the critical stress intensity factor, K_{Ic}, and is the material property used in this method. The K_{Ic} of the material is either directly measured as a function of temperature, or is conservatively estimated, using information from other fracture toughness tests.

b. The Code specifies the maximum K_{Ic}, as a function of temperature, that can be assumed for the specific material, based on results of tests on the material used. This value is called K_{IR}, reference stress intensity factor. The Code also provides rules for calculating the K_I, including definitions of postulated flaws, and specifies the safety factors to be applied. The acceptance criterion is that the K_{IR} of the material must always be higher than the K_I calculated.

c. Direct measurement of the K_{Ic} as a function of temperature is expensive and time consuming and requires more sample material than is usually available. Correlations between the K_{Ic} determined directly and results of simpler fracture toughness tests are not exact, but may be used if appropriate allowances are made for variations in material behavior and data scatter. The Code gives values of K_{IR} as a function of temperature relative to a conservative determination of the nil-ductility transition temperature (NDTT) of the material. This reference temperature, RT_{NDT}, is determined for the ferritic materials of components for which operating and testing limit curves must be calculated. The effects of radiation on the fracture toughness of the material in the beltline region of the reactor vessel is accounted for by adjusting the RT_{NDT} of the affected material upward. The amount of upward shift depends on the composition of the steel (especially its copper and phosphorous content), the neutron fluence, and the temperature of irradiation. Conservative predictions of the effect of radiation on the RT_{NDT} based on data in the literature are factored into the original limit curves. The continued conservatism of these predictions throughout plant life is verified by a mandatory material surveillance program described in Appendix H to 10 CFR Part 50.

d. The Code specifies the stress components that must be used for the K_I calculations, and the factors that must be applied to each to provide adequate safety margins. The Code, by reference to WRC-175, specifies the expression to use for calculating the K_I, using the applied stresses and the postulated flaw geometry. Although calculations are usually made by a computer, curves are provided in the Code to facilitate the use of conservative hand calculations if desired.

3. Pressure-Temperature Requirements

The requirements for the pressure-temperature limits are as follows:

a. Pressure-Temperature Limits for Preservice Hydrostatic Tests

During preservice hydrostatic tests (if fuel is not in the vessel), the K_{IR} must be greater than the K_I caused by pressure. The expression used is:

$$K_I = K_I(\text{pressure}) < K_{IR}$$

b. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests

During performance of inservice leak and hydrostatic tests, the K_{IR} must be greater than 1.5 times the K_I caused by pressure. The expression used is:

$$K_I = 1.5\ K_I(\text{pressure}) < K_{IR}$$

c. Pressure-Temperature Limits for Heatup and Cooldown Operations

At all times during heatup and cooldown operations, the K_{IR} must be greater than the sum of 2 times tne K_I caused by pressure and the K_I caused by thermal gradients. The expression used is:

$$K_I = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{IR}$$

d. Pressure-Temperature Limits for Core Operation

At all times that the reactor core is critical (except for low power physics tests) the temperature must be higher than that required for inservice hydrostatic testing, and in addition, the pressure-temperature relationship shall provide at least a 40°F margin over that required for heatup and cooldown operations.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. Preliminary Safety Analysis Report (PSAR)

Actual operating limit curves cannot be determined at the PSAR stage, because the fracture toughness and other required tests have not been performed on the actual material that will be used. Typical curves, with temperatures shown relative to the RT_{NDT}, and the basis for determining the curves are reviewed and compared with the acceptance criteria described in II above.

2. Final Safety Analysis Report (FSAR)

The limits in the plant Technical Specifications will be shown using real temperature. These curves and their bases are reviewed to determine acceptability in the following areas:

a. The limiting RT_{NDT} has been properly determined, and radiation effects are included in a conservative manner.
b. Limits are shown for all required conditions.
c. The limits proposed are consistent with the acceptance criteria described in II above.
d. The procedures for updating the limit curves, in conjunction with scheduled tests on material surveillance specimens, are well defined and included in the Technical Specifications.

3. Acceptability Determination Methods

The reviewer evaluates each limit curve for acceptability by performing check calculations using the simplified methods referenced in the Code and WRC Bulletin 175 that have been verified by the Materials Engineering Branch to yield conservative

values. These methods are described in detail by examples below, and the curves necessary to perform the calculations are included herein as Figures 1, 2 and 3. Figure 4 is an example of acceptable limit curves developed by this method.

a. Preservice Hydrostatic Tests

The preservice hydrotest at 1.25 design pressure corresponds to the standard Code component hydrotest usually performed in the shop, but in this case it is the hydrotest for field welds, so it may involve the entire reactor coolant system.

The Code recommends that component hydrostatic tests be run at a temperature no lower than RT_{NDT} + 60°F, but also recommends that system tests should have more stringent requirements. The MTEB position is that the minimum temperature for the preservice test, if fuel is not in the vessel, be determined using the methods of Code Section III, Appendix G, using less stringent factors.

First, the RT_{NDT} of the vessel material must be determined. This is defined by the Code for new plants, and is essentially a conservative value of the NDTT as determined by drop weight test. Guidelines for estimating the RT_{NDT} if the prescribed tests have not been run are covered by Branch Technical Position - MTEB No. 5-2, "Fracture Toughness Requirements."

The reference temperature and the toughness of the material at any temperature is a function of the difference between the RT_{NDT} of the material and the temperature of interest. The Code provides a curve (Figure G-2110.1) for the allowable calculated stress intensity factor (K_{IR}) as a function of the temperature relative to RT_{NDT}.

The Code also provides a recommended basis for calculating K_I, including recommendations for assumed flaw size and shape, and appropriate front and back surface correction factors. Because the assumed flaw size is proportional to the wall thickness, t (flaw depth = 0.25 t and length = 1.5 t), the K_I expressions are simplified to multipliers that are a function only of wall thickness and stress level. These factors, M_m for membrane stresses and M_B for bending stresses, are provided in graphical form in Figure G-2114.1.

The criterion recommended by MTEB can be expressed as

$K_I < K_{IR}$ for the shell region.

To get K_I, the stress level and wall thickness must be known. The pressure for the hydrostatic test is 1.25 times the design pressure, so either of two simple methods can be used to approximate the membrane stress accurately enough for this purpose:

stress = 1.25 times the Code allowable (S_m)

stress = $\frac{Pr}{t}$

where P is the test pressure and r is the vessel radius. As an example, assume a vessel with a design pressure of 2500 psig, made of steel with an S_m

of 26,700 psi, and a minimum yield strength of 50,000 psi. The stress for the preservice hydrotest is then

$26{,}700 \times 1.25 = 33{,}400$ psi, or

$\frac{(1.25)\ (2500)\ (95)}{9} = 33{,}400$ psi, for a vessel with a radius of 95 inches and a wall thickness of 9 inches.

The next step is to determine the factor to apply to this stress to obtain K_I. Figure G-2114.1 (reproduced here as Fig. 1) provides several curves, depending on the ratio of the stress level to the yield strength of the material. In this case, the stress level is 33,400; the yield strength is conservatively assumed to be 50,000 so the curve for a ratio of .7 should be used. (A ratio equal to or higher than the actual ratio must be used for conservatism.) For a 9-in. thick vessel ($\sqrt{t} = 3$), the value of M_m from Figure G-2114.1 is 2.94. The K_I for this case is then:

$K_I = (M_m)$ (Membrane Stress)

$K_I = (2.94)\ (33{,}400) = 98{,}300$ psi $\sqrt{\text{in.}}$

From Figure G-2110.1 (reproduced here as Fig. 2), a temperature of at least RT_{NDT} + 120°F is necessary for a K_I of this level.

If, for example, an original RT_{NDT} of 40°F is assumed, the required temperature is then 40 + 120, or 160°F.

b. Inservice Leak and Hydrotest

The temperatures for the inservice leak and hydrotest, performed at operating pressure and about 1.1 operating pressure, respectively, are calculated in essentially the same way. The differences are that a factor of 1.5 must be applied to the calculated K_I to provide extra margin, and the stress levels are lower, so the value of M_m is taken from a lower ratio curve.

Using the same vessel as an example, with a normal operating pressure (P_o) of 2250 psi, the membrane stress for the leak test can be approximated as:

$\frac{\text{operating pressure}}{\text{design pressure}} \times$ allowable stress

or $\frac{2250}{2500} \times 26{,}700 = 24{,}000$ psi

This is about half of the minimum yield strength, so the M_m is taken from the 0.5 ratio curve, and is 2.87. The calculated K_I that must be assumed is then:

$K_I = (1.5)\ (M_m)$ (Membrane Stress)

or $K_I = (1.5)\ (2.87)\ (24{,}000) = 103{,}500$ psi $\sqrt{\text{in.}}$

From the K_{IR} curve, a temperature of about RT_{NDT} + 125°F is required. As this is an inservice test, the RT_{NDT} would probably have been increased from its original value of + 40°F by some shift caused by radiation. Assume this shift is 100°F, thus the temperature for the leak test must be at least:

40 + 100 + 125 = 265°F

The inservice hydrotest temperature (at 1.1 P_o) is determined in exactly the same way, and requires a minimum temperature of about RT_{NDT} + 133°F, or 273°F.

c. Heatup, Cooldown, and Normal Operation

For normal operation, which includes upset conditions and startup and shutdown procedures, operating limit curves must be provided that show the maximum permissible pressure at any temperature from cold shutdown conditions to full pressurization conditions.

Reactor vendors have developed computer codes to perform the necessary calculations, because thermal stresses must be included, and hand calculations of even moderate sophistication are very time consuming. WRC Bulletin 175 includes a set of curves derived from computer programs that can be used to approximate the K_I caused by thermal stresses, as a function of wall thickness and rate of temperature change. Pressure-temperature curves developed using these approximations agree fairly well with those determined using much more rigorous procedures, and can be used with confidence to evaluate the proposed operating limits given in Technical Specifications. These curves require the calculation of only 3 to 5 points. Either allowable pressure at a given temperature, or allowable temperature at a given pressure can be calculated. It is usually more convenient to calculate allowable minimum temperature, so this method will be used in the example.

Using the same reactor vessel as in the previous example, and a rate of temperature change of 50°F per hour, calculations of required temperatures for several pressures are illustrated. The curves for thermal effects given in WRC Bulletin 175 are very conservative, thus no additional margin need be applied to the K_I from thermal stress, but a factor of 2.0 is used on primary stresses. The basic expression is then:

$$K_{IR} \geq 2\ K_I(\text{membrane}) + K_I\ (\text{thermal})$$

K_I(membrane) is calculated exactly as in the previous examples. K_I(thermal) for a 9-in. thick wall, at 50°/hr is about 12,000 psi $\sqrt{\text{in.}}$ from Figure 4-5, WRC Bulletin 175 (reproduced here as Fig. 3).

Thus, for a pressure of 2250 psig, a membrane stress of 24,000 psi, and M_m of 2.87, the basic expression is given by

$$K_{IR} > (2)(24{,}000)(2.87) + 12{,}000 = 150{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the K_{IR} curve, a temperature of RT_{NDT} + 158°F is required. With an RT_{NDT} of 140°F, the temperature required for operating pressure at a heatup or cooldown rate of 50°/hr is then

$$140 + 158 = 298°F$$

For a pressure of 1/2 of operating (1125 psig), the membrane stress is 1/2 of that at operating pressure, or 12,000 psi.

The M_m can be taken from the 0.5 $\frac{\sigma}{\sigma_y}$ ratio curve in Figure 2114.1, so is again 2.87.

$$K_{IR} \geq (2)(12{,}000)(2.87) + 12{,}000 = 81{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the K_{IR} curve, the minimum temperature is RT_{NDT} + 100°F, or 140 + 100 = 240°F.

The same calculation for a pressure of 1/5 operating pressure (450 psig and 4800 psi stress) is similar, but in this case the stress is less than .1 of the yield strength, so the M_m (from the .1 ratio curve) is only 2.82.

$$K_{IR} \geq (2)(4800)(2.82) + 12{,}000 = 39{,}000 \text{ psi } \sqrt{\text{in.}}$$

The K_{IR} curve shows that the minimum temperature is RT_{NDT} + 0°F, or 140°F.

Three points on a 50°/hr operating limit curve for this vessel at this time in its service lifetime have thus been calculated:

Pressure (psig)	Min. Temperature (Fahrenheit)
450	140
1150	240
2250	298

A smooth curve drawn through these points will very closely approximate the results using more rigorous methods.

d. Core Operation

Appendix G, 10 CFR Part 50, specifies pressure-temperature limits for core operation to provide additional margin during actual power production.

The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum pressure-temperature curve for heatup and cooldown calculated as described in the preceding section. The minimum temperature for the inservice hydrostatic test for the vessel used in the preceding example was 273°F. A vertical line at 273°F on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve as determined in the preceding section, constitutes the limit for core operation for this example.

The information required to evaluate the adequacy of the temperature limits can all be put on one figure. The temperature limits calculated in the preceding sections along with material data used are shown in Figure 4.

e. Acceptable limit curves for several typical plants have been developed to facilitate the evaluation of acceptability of proposed limits. These are included as Figures 5 and 6 of this review plan. These are based on the simplified procedures described above, and are slightly more conservative than curves developed by more rigorous computer calculations. Curves presented in plant Technical Specifications are considered acceptable if they are as conservative as these reference limit curves. If they are based on more rigorous analytical methods, as recommended by the Code or WRC Bulletin 175, they will be considered acceptable if the variation from the reference limit curves is not more than 10°F.

If the proposed limit curves are more than 10°F less conservative than the reference limit curves, detailed bases and calculations must be submitted by the applicant for review. To be acceptable, all bases and analytical expressions used must be in accordance with Appendix G, 10 CFR Part 50, and the proposed curves must agree with check calculations made by the Materials Engineering Branch using these bases and expressions.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and that the completeness and technical adequacy of his evaluation will support the following kind of concluding statement, to be included in the staff's safety evaluation report:

"The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to assure adequate safety margins against nonductile or rapidly propagating failure are in conformance with established criteria, codes, and standards acceptable to the Regulatory staff. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 31."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

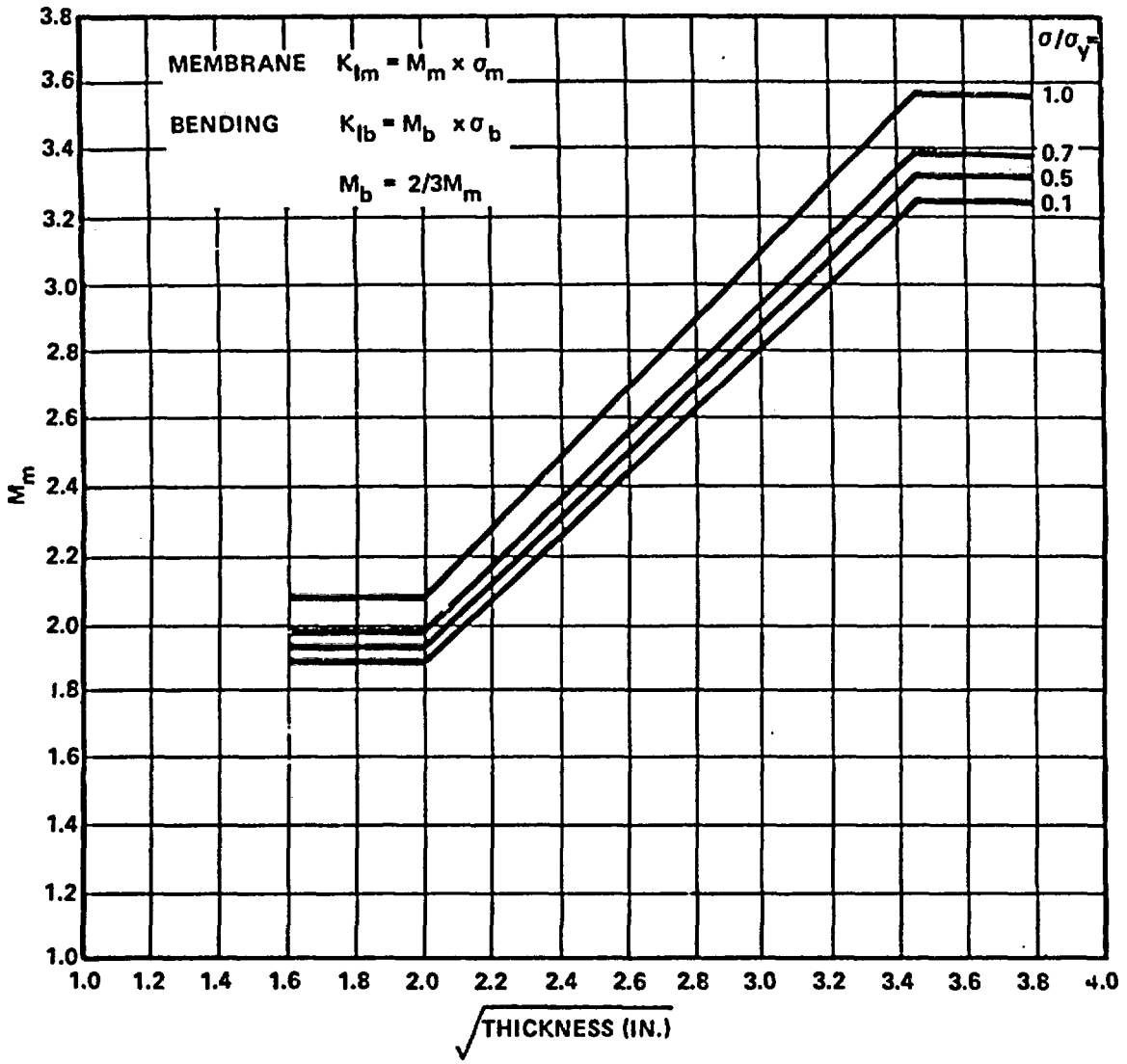
2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4. ASME Boiler and Pressure Vessel Code, Section III, including Appendix G, "Protection Against Nonductile Failure," American Society of Mechanical Engineers.

5. WRC Bulletin 175, "PVRC Recommendation on Fracture Toughness," Welding Research Council.

6. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants," appended.



M_m AND M_b VS. WALL THICKNESS FOR

SEMI-ELLIPTICAL SURFACE FLAW ¼T DEEP AND 1½T LONG

FIGURE 1

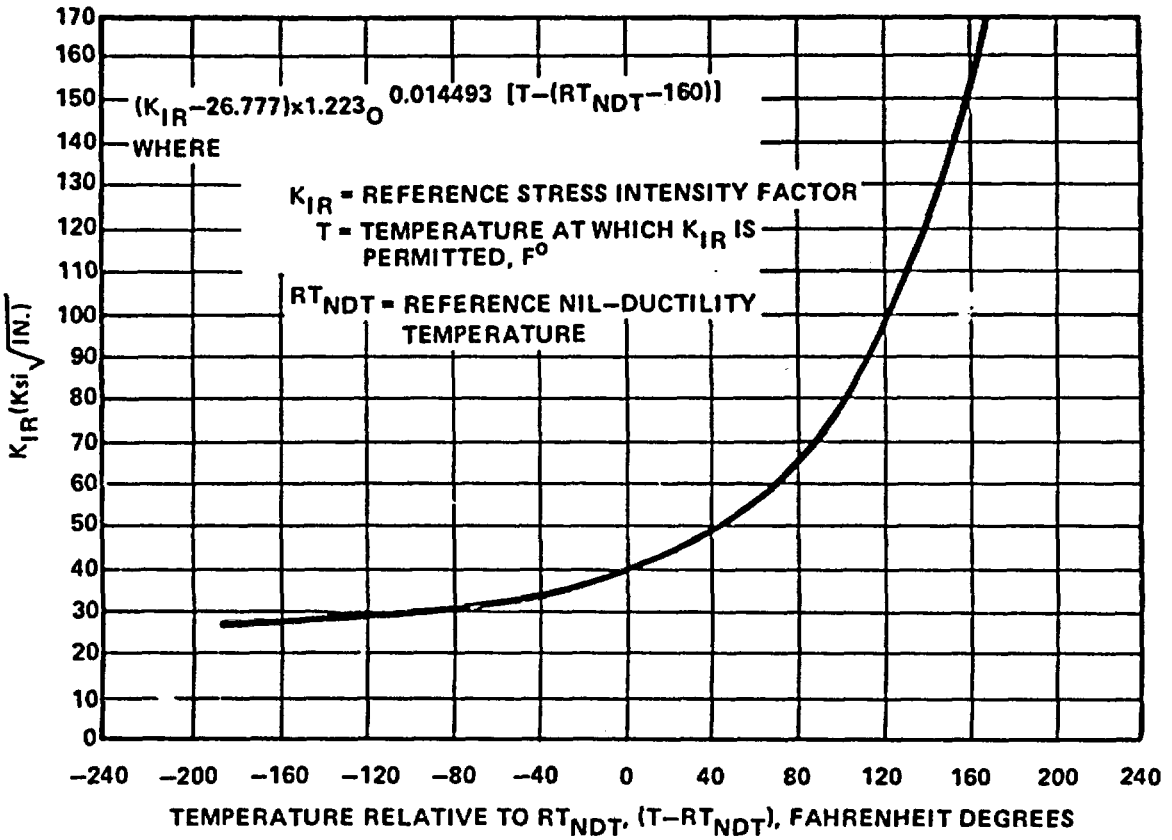
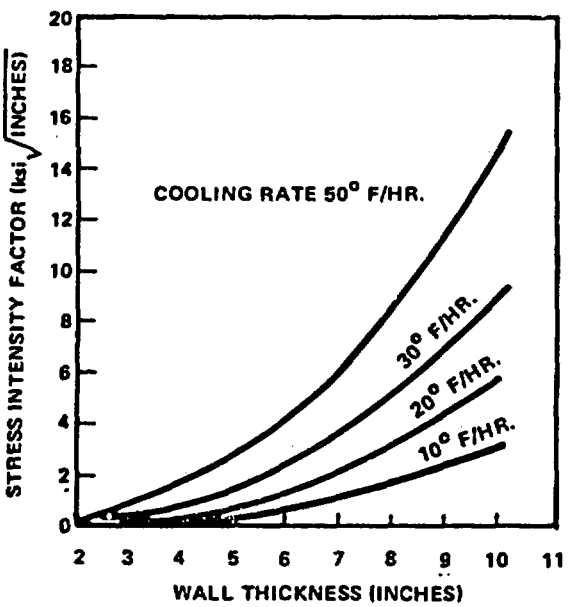


FIGURE 2



4-5 – STRESS INTENSITY FACTOR CAUSED BY THERMAL STRESS FOR CYLINDERS WITH RADIUS/THICKNESS = 10

FIGURE 3

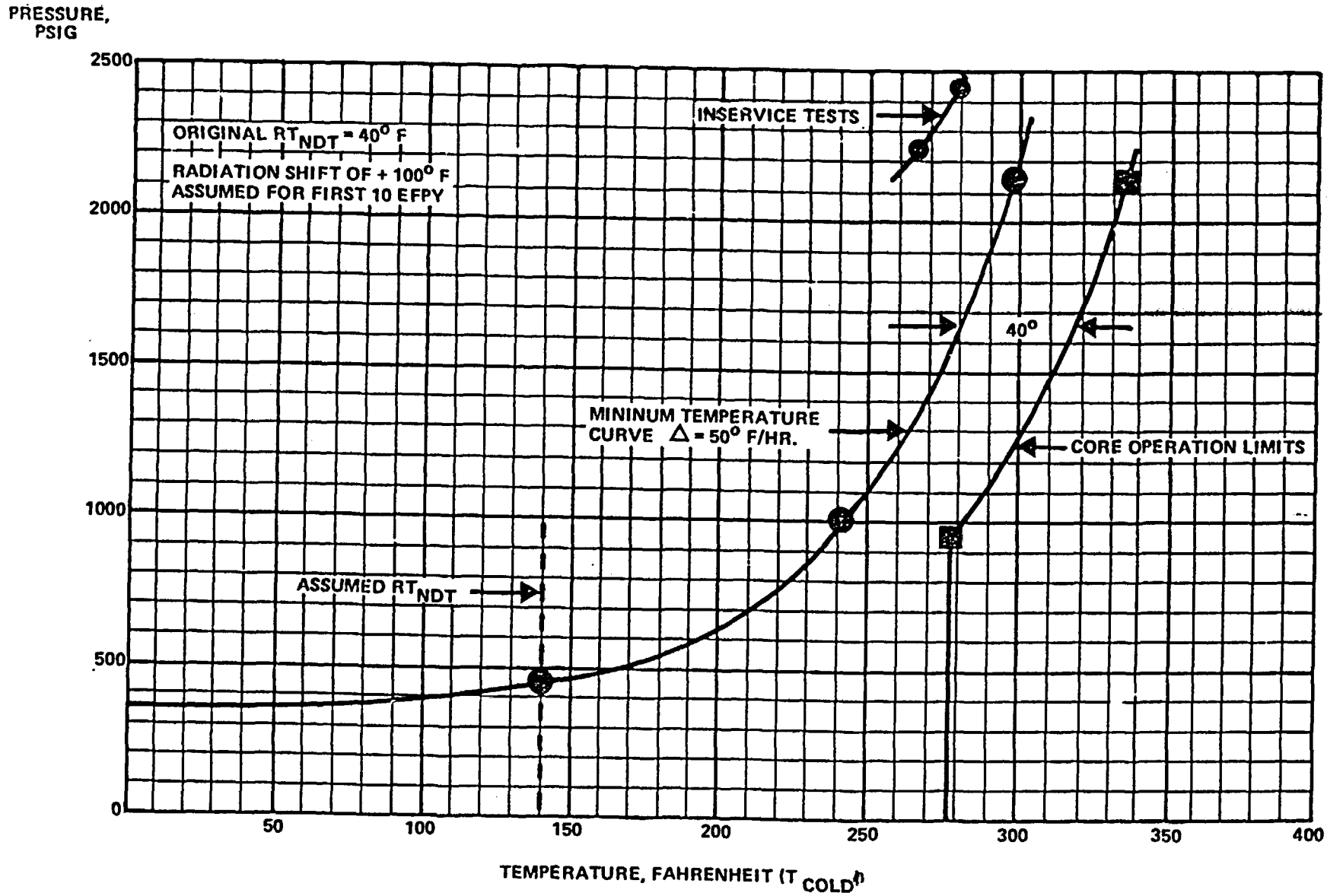


FIGURE 4 TYPICAL PWR REACTOR VESSEL TEMPERATURE LIMITS

5.3.2-11

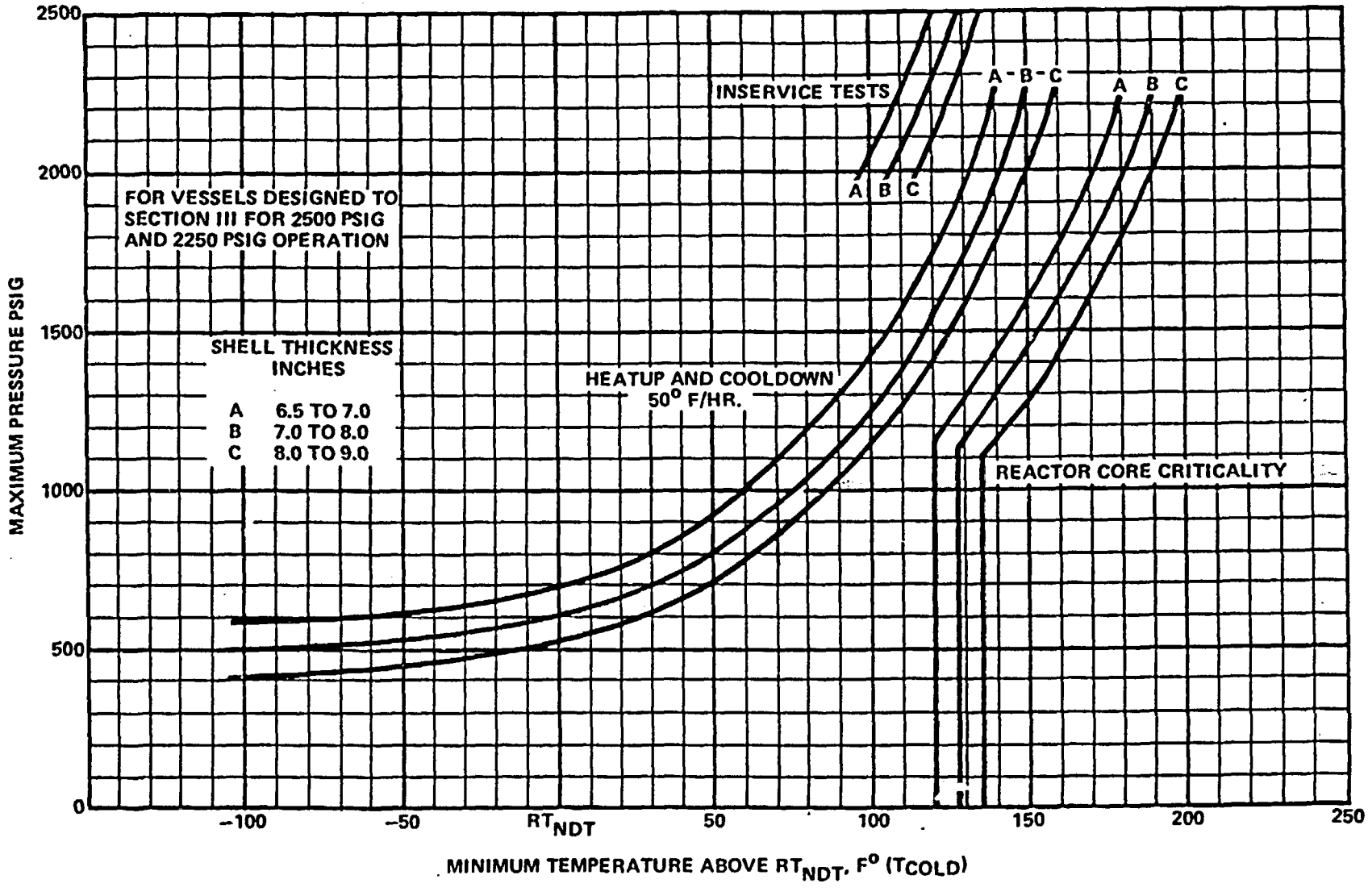
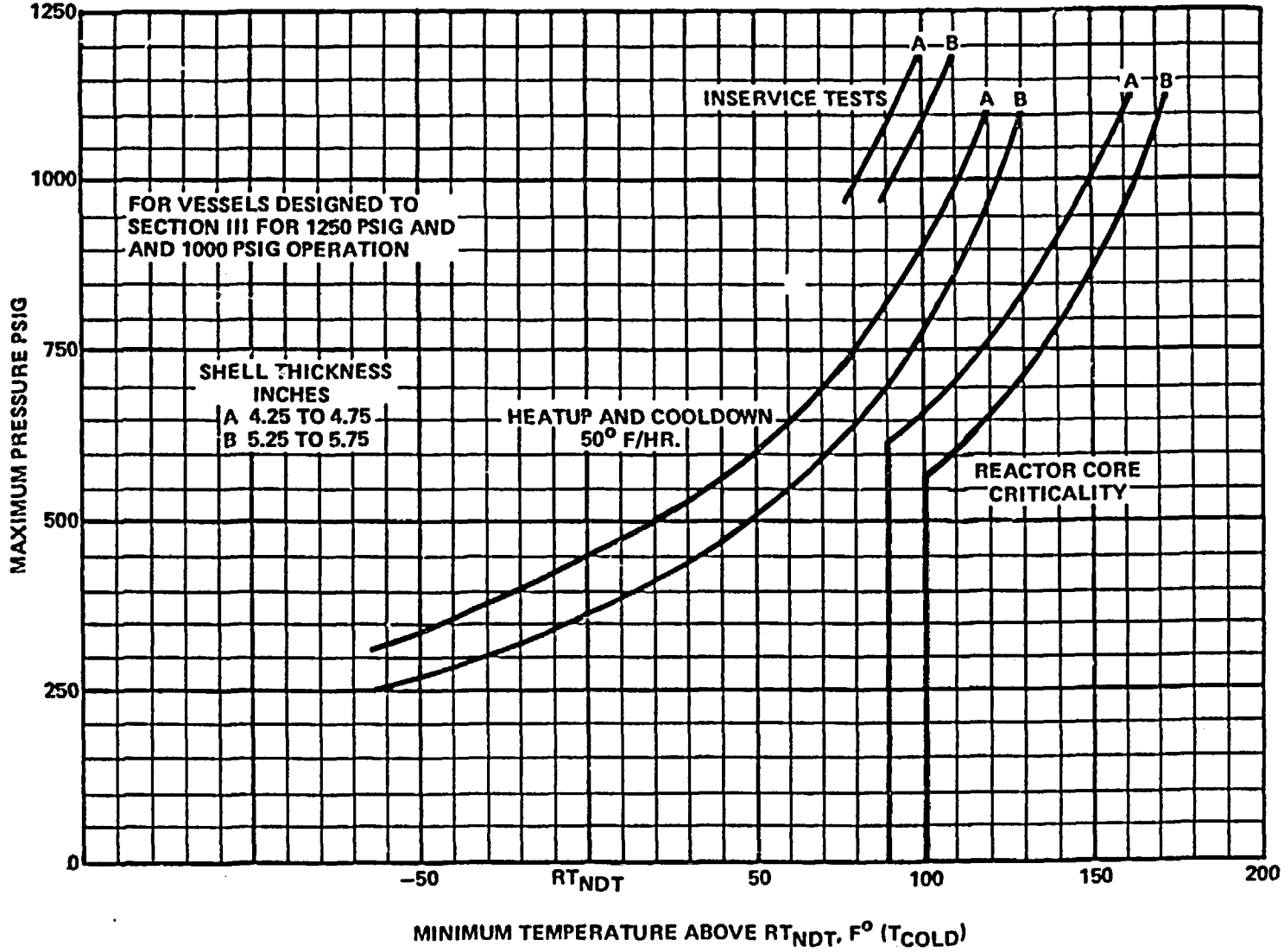


FIGURE 5 PWR REFERENCE LIMIT CURVES

5.3.2-12



5.3.2-13

FIGURE 6 BWR REFERENCE LIMIT CURVES

BRANCH TECHNICAL POSITION - MTEB NO. 5-2
FRACTURE TOUGHNESS REQUIREMENTS

A. Background

Current requirements regarding fracture toughness, pressure-temperature limits, and material surveillance are covered by the ASME Code and Appendices A, G, and H to 10 CFR Part 50. The purpose of this branch technical position is to summarize these requirements and provide clarification, as necessary.

Since many of these requirements were not in force when the plant was designed and built, this position also provides guidance for applying these requirements to older plants. Also included is a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with the new requirements.

B. Branch Technical Position

1. Preservice Fracture Toughness Test Requirements.

The fracture toughness of all ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary shall be evaluated in accordance with the requirements of Section III of the ASME Code, as augmented by Appendix G, 10 CFR 50. The fracture toughness test requirements for plants with construction permits prior to August 15, 1973 may not comply with the new Codes and Regulations in all respects. The fracture toughness of the materials for these plants must be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This must be done because the operating limitations imposed on old plants must provide the same safety margins as are required for new plants.

1.1 Determination of RT_{NDT} for Vessel Materials

Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from results of fracture toughness tests. Both drop weight NDTT tests and Charpy V-notch tests must be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208) is the RT_{NDT} if, at 60°F above the NDTT, at least 50 ft-lbs of energy and 35 mils lateral expansion are obtained in Charpy V tests on specimens oriented in the weak direction (traverse to the direction of maximum working).

In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests required to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided for guidance.

(1) If dropweight tests were not performed, but full Charpy V-notch curves were obtained, the NDTT for SA-533 Grade B, Class 1 plate and weld material may be assumed to be the temperature at which 30 ft-lbs was obtained in Charpy V-notch tests, or 0°F, whichever was higher.

(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures:

(a) 60°F.

(b) The temperatures of the Charpy V-notch upper shelf.

(c) The temperature at which 100 ft-lbs was obtained on Charpy V-notch tests if the upper-shelf energy values were above 100 ft-lbs.

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 50 ft-lbs and 35 mils LE would have been obtained on traverse specimens may be estimated by one of the following criteria:

(a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.

(b) Temperatures at which 50 ft-lbs and 35 mils LE were obtained on longitudinally-oriented specimens increased 20°F to provide a conservative estimate of the temperature that would have been required to obtain the same values on transversely-oriented specimens.

(4) If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 30 ft-lbs was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 45 ft-lbs was obtained if the specimens were longitudinally-oriented. If the minimum value obtained was less than 45 ft-lbs, the RT_{NDT} may be estimated as 20°F above the test temperature.

1.2 Estimation of Charpy V Upper-Shelf Energies

For the beltline region of reactor vessels, the upper shelf toughness must be adequate to accommodate degradation by neutron radiation. The original minimum shelf energy must be 75 ft-lbs for vessels with an estimated end of life neutron fluence (> 1 MeV) of 1×10^{19} and over. A value of 70 ft-lbs is considered adequate for material for vessels that will be subjected to lower fluences.

If upper-shelf Charpy energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed.

If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties.

1.3 Reporting Requirements

Fracture toughness information required by the Code and by Appendix G, 10 CFR 50, must be reported in the FSAR to provide a basis for evaluating the adequacy of the operating limitations given in the Technical Specifications. In the case of older plants, the data may be estimated, using the procedures listed above, or other methods that can be shown to be conservative.

2. Operating Limitations for Fracture Toughness

2.1 Required Pressure-Temperature Operating Limitations

As required by Appendix G, 10 CFR 50, the following operating limitations shall be determined and included in the Technical Specifications. The basis for determination

shall be reported, and is the responsibility of the applicant, but in no case shall the limitations provide less safety margin than those determined in accordance with Appendix G, 10 CFR 50, and Appendix G to Section III of the Code.

(1) Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system.

(2) Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service.

(3) Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions.

(4) Maximum pressure-minimum temperature curves for core operation.

2.2 Recommended Bases for Operating Limitations

2.2.1 Leak and Hydrostatic Tests

(1) It is recommended that no tests at pressures higher than design pressure be conducted with fuel in the vessel.

(2) Tests at pressures less than design pressure should be conducted at temperatures calculated according to Appendix G of Section III of the Code for the beltline region (including conservative estimates of radiation damage, see Section 3.0 below) if the maximum calculated primary stress in no other region of the vessel exceeds 1.25 S_m during the test, and the RT_{NDT} of the beltline is assumed to be at least 30°F above that of the higher stressed regions. If primary stresses are calculated to be over 1.25 S_m in any region during the test, the RT_{NDT} of the vessel must be assumed to be at least 50°F higher than that of any region where the calculated primary stresses are over 1.25 S_m.

(3) Alternatively, a fracture mechanics analysis, with technical justification for all assumptions and bases, may be made to determine the minimum test temperature. In no event shall the minimum temperature be lower than that resulting from calculations for the beltline region in accordance with Appendix G of the Code.

2.2.2 Heatup and Cooldown Limit Curves

Heatup and cooldown pressure-temperature limit curves may be determined using simple $\frac{pr}{t}$ stress calculations, using the method given in Appendix G of the Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the Code or from Figure 4-5 in WRC Bulletin 175.

Calculations need only be performed for the beltline region, if the assumed RT_{NDT} of the beltline is at least 50°F above the RT_{NDT} for all higher stressed regions.

Alternatively, more rigorous analytical procedures may be used, provided that the intent of the Code is met, and adequate technical justification for all assumptions and bases is provided.

2.2.3 Core Operation Limits

To provide added margins during actual core operation, Appendix G, 10 CFR 50 requires a minimum temperature during core operation, and a 40°F margin in temperature over the pressure-temperature limits as determined for heatup and cooldown in 2.2.2 above. The minimum temperature, regardless of pressure, is the temperature calculated for the inservice hydrostatic test according to 2.2.1 above.

2.2.4 Upset Conditions

The pressure-temperature limits described in 2.2.2 and 2.2.3 above are applicable to upset conditions. Normal operating procedures must permit variations from intended operation, including all upset conditions, without exceeding the limit curves.

2.2.5 Emergency and Faulted Conditions

It is recognized that the severity of a transient resulting from an emergency or faulted condition is not directly related to operating conditions, and resulting temperature-stress relationships in the reactor coolant boundary components are primarily system dependent, and therefore not under direct control of the operator.

For these reasons, operating limits for emergency and faulted conditions are not a requirement of the Technical Specifications.

The SAR must present evaluations of the continued integrity of all vital components during postulated faulted conditions. It is recommended that such evaluations be made in a realistic manner, avoiding grossly overconservative assumptions and procedures, and clearly show that margins against loss of integrity are calculable and adequate.

2.3 Reporting Requirements

The Technical Specifications must include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications must also include information on the intended operating procedures, and justify that adequate margins between the expected conditions and the limit conditions will be provided to protect against unexpected or upset conditions.

3. Inservice Surveillance of Fracture Toughness

The reactor vessel may be exposed to significant neutron radiation during the service life. This will affect both the tensile and toughness properties. A material surveillance program in conformance with Appendix H, 10 CFR 50, must be carried out.

3.1 Surveillance Program Requirements

The minimum requirements for the surveillance program are covered by Appendix H, 10 CFR 50. It is strongly recommended that consideration by given to the desirability of additional surveillance methods, such as the inclusion of CT, DWT, DT, or other specimens to provide the capability of redundant test methods and analytical

procedures, particularly if the estimated neutron fluence is over 2×10^{19}, or the toughness of the vessel material is marginal.

The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-73, unless the intent of the program is better realized by using more rigorous criteria. For example, the approach of estimating the actual RT_{NDT} and upper shelf toughness of each plate, forging, or weld in the beltline as a function of service life, and choosing as the surveillance materials those that are expected to be most limiting, may be preferable in some cases. This would include consideration of the initial RT_{NDT}, the upper shelf toughness, the expected radiation sensitivity of the material (based on copper and phosphorous content, for example) and the neutron fluence expected at its location in the vessel.

3.2 SAR Requirements

The adequacy of the surveillance program cannot be evaluated unless all pertinent information is included in the SAR. Information requested for beltline materials includes the following:

(1) Tensile properties.
(2) DWT and Charpy V test results used to determine RT_{NDT}.
(3) Charpy V test results to determine the upper shelf toughness.
(4) Composition, specifically the copper and phosphorous content.
(5) Estimated maximum fluence for each beltline material.
(6) List of materials included in the surveillance program, with basis used for their selection.

3.3 Surveillance Test Procedures

Surveillance capsules must be removed and tested at intervals in accordance with Appendix H, 10 CFR 50. The proposed removal and test schedule shall be included in the Technical Specifications.

3.4 Reporting Requirements

All information used to evaluate results of the tests on surveillance materials, evaluation methods, and results of the evaluation should be submitted with the evaluation report. This should include:

(1) Original properties and compositions of the materials.
(2) Fluence calculations, including original predictions, for both surveillance specimens and vessel wall.
(3) Test results on surveillance specimens.
(4) Basis for evaluation of changes in RT_{NDT} and upper shelf toughness.
(5) Updated prediction of vessel properties.

3.5 Technical Specification Changes

Changes in the operating and test limits recommended as a result of evaluating the properties of the surveillance material, together with the basis for these changes, shall be submitted to the Directorate of Licensing for approval.

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SECTION 5.3.3 REACTOR VESSEL INTEGRITY

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

The portions of the applicant's safety analysis report (SAR) listed below are reviewed. These portions are all related to the integrity of the reactor vessel. Although most of these areas are reviewed separately in accordance with other review plans, the integrity of the reactor vessel is of such importance that a special summary review of all factors relating to the integrity of the reactor vessel is warranted. The information in each area is reviewed to ensure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

1. Design

   The basic design of the reactor vessel is reviewed.

2. Materials of Construction

   The materials of construction are each taken into consideration.

3. Fabrication Methods

   The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are reviewed.

4. Inspection Requirements

   The inspection test methods and requirements are reviewed.

5. Shipment and Installation

   Protective measures taken during shipment of the reactor vessel and its installation at the site are reviewed.

6. Operating Conditions

   All the operating conditions as they relate to the integrity of the reactor vessel are reviewed.

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

7. Inservice Surveillance

Plans and provisions for inservice surveillance of the reactor vessel are reviewed.

II. ACCEPTANCE CRITERIA

The basic acceptance criteria for each review area are covered by other standard review plans, so they will be discussed here only in general terms. References are made to the review plans that include detailed criteria. Interrelationships among review areas, and criteria for consistency, compatability, and technical coherence among review areas, are emphasized in the following discussion.

1. Design

The basic acceptance criteria for the design of the vessel are detailed in the standard review plans for SAR Sections 3 and 5.2. These cover the requirements of the General Design Criteria and Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"). The design of the reactor vessel must be compatible with the properties of the materials used, and must permit construction by the use of standard and well proven fabrication methods. The design details should not include new or novel concepts unless they are substantiated by a comprehensive justification showing that no aspects of the design will compromise the overall integrity of the vessel in any manner.

The design details must be adequate to permit all required inspections and to provide required access to all areas requiring inservice inspection in conformance with Section XI of the Code, as detailed in Standard Review Plan 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

If the neutron radiation exposure of the reactor vessel becomes high enough that the predicted value of the adjusted reference temperature of the material exceeds 200°F, the design must be adequate to permit in-place annealing of the vessel to restore ductility and toughness, in accordance with Appendix G, 10 CFR Part 50, and as detailed in the review plan for SAR Section 5.3.1, "Reactor Vessel Materials."

2. Materials of Construction

The basic acceptance criteria for the materials used in the construction of the reactor vessel are the requirements of Section III of the Code, as augmented by Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements." These criteria are detailed in Standard Review Plans 5.2.3, "Reactor Coolant Pressure Boundary Materials," and 5.3.1, "Reactor Vessel Materials."

The materials must be compatible with the design requirements. Acceptability is based on standard practice and engineering judgement, with consideration being given to such factors as material form, size-related variations in properties, and nonisotropic characteristics.

Although many materials are acceptable for reactor vessels according to Section III of the Code, the special considerations relating to fracture toughness and radiation

effects effectively limit the basic materials that are currently acceptable for most parts of reactor vessels to SA 533 Gr B Cl 1, SA 508 Cl 2, and SA 508 Cl 3. Acceptability criteria for other grades will have to be developed before they can be used.

The relationships among material compositions, expected neutron fluence, and requirements for the material surveillance program must be compatible. The reviewer uses published data to ensure that the predicted shift in toughness properties (RT_{NDT} and upper shelf energy) is conservative, based on actual material composition and predicted fluence. Acceptability of the material surveillance program, as specified in Appendix H, 10 CFR Part 50, depends on these relationships.

3. Fabrication Methods

Acceptability criteria for the basic fabrication processes and their qualification and control requirements are given in Sections III and IX of the Code, and detailed in Standard Review Plan 5.3.1, "Reactor Vessel Materials."

Although a particular fabrication process (such as multiple wire-high heat input welding) may be generally acceptable, it may not be suitable for reactor vessel fabrication for some materials without further justification or qualification. The reviewer uses "state-of-the-art" criteria and past practice to evaluate the acceptability of materials-process combinations.

Because fabrication methods, materials, and the effectiveness of non-destructive evaluation methods are interrelated, the reviewer must rely on state-of-the-art knowledge and past practice to determine whether the proposed combinations are compatible and acceptable.

4. Inspection Requirements

The basic requirements for performing nondestructive inspections and the quality assurance criteria for the reactor vessel are contained in Sections III and V of the Code. These are detailed in Standard Review Plan 5.3.1, "Reactor Vessel Materials."

Acceptance criteria for compatibility with materials and fabrication areas are discussed in previous sections.

Very important relationships are those among in-process and final shop inspections, and the inservice inspection requirements of Section XI of the Code. The reviewer must determine that the methods of inspection, the sensitivity levels, and flaw evaluation criteria are compatible with Section XI, and that the results of the preservice baseline inspection can be correlated with the results of later inservice inspections.

5. Shipment and Installation

The basic acceptance criteria for procedures and care used in shipping, storage, and installation of the vessel are given in Regulatory Guides 1.37, "Quality Assurance

Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Reactor Plants," 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," and 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

The purpose of this area of review is to verify that the as-built characteristics of the reactor vessel are not degraded by improper handling. Acceptability in these areas is assured for current designs and materials by compliance with the basic acceptance criteria. If nonstandard materials or designs are used, the reviewer must determine that these criteria will be adequate, based on current technology.

If the basic criteria are not followed, either intentionally or through error, the reviewer must evaluate, on a case basis, whether the integrity of the reactor vessel is compromised, using current technology, past practice, and experience as applicable.

6. Operating Conditions

Acceptance criteria for operating limits for the reactor vessel are given in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," and are detailed in Standard Review Plan 5.3.2, "Pressure-Temperature Limits." In addition, Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" provides acceptable criteria for other phases of operational procedures.

Abnormal operational occurrences must not result in loss of reactor vessel integrity. The most severe postulated transient is the thermal shock to the vessel caused by emergency core cooling system operation after a loss-of-coolant accident. The criterion for acceptable behavior is that the vessel must remain leaktight enough to support adequate core cooling. The generally accepted principles and procedures of linear elastic fracture mechanics provide the basis for acceptance of analyses that support conformance with this criterion.

7. Inservice Surveillance

The acceptance criteria for adequacy of the reactor vessel materials surveillance program are based on the requirements of Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," and detailed in Standard Review Plan 5.3.1, "Reactor Vessel Materials."

The SAR also provides information regarding the inservice inspections to be performed on the reactor vessel. The acceptance criteria for accessibility and inspection plan details are those of Section XI of the Code, and are detailed in Standard Review Plan 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case. The reviewer initially determines that the basic criteria are met in each review area covered by this plan. Although he will not normally

be responsible for the basic reviews of all of these areas, he will consult with those responsible for basic review of the other areas to determine that all areas are individually acceptable.

He then reviews each area again, considering the information presented in other areas that interrelate with it, as discussed in II above.

Because the reviewer is familiar with the specific procedures used by the reactor vendor, he can readily pick out any differences from past practice. He will evaluate these in detail, consulting with other MTEB members as appropriate.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided to satisfy the requirements of this review plan, and that the completeness and technical adequacy of his evaluation will support conclusions of the following type, to be included in the staff's safety evaluation report:

"We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for this plant. The bases for our conclusion are that the design, materials, fabrication, inspection, and quality assurance requirements for the plant will conform to applicable AEC Regulations and Regulatory Guides, and to the rules of the ASME Boiler and Pressure Vessel Code, Section III. The stringent fracture toughness requirements of the Regulations and ASME Code Section III will be met, including requirements for surveillance of vessel material properties throughout service life. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Non-Ductile Failure," of ASME Code Section III, and Appendix G, 10 CFR Part 50.

"The integrity of the reactor vessel is assured because the vessel

(1) will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and any pertinent Code Cases;
(2) will be made from materials of controlled and demonstrated high quality;
(3) will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies;
(4) will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation, and that the vessel will not fail under the conditions of any of the postulated accidents;
(5) will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions; and
(6) may be annealed to restore the material toughness properties if this becomes necessary."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4. ASME Boiler and Pressure Vessel Code, Section III, especially Appendix G, "Protection Against Nonductile Failure," American Society of Mechanical Engineers.

5. ASME Boiler and Pressure Vessel Code, Sections II, V, IX, and XI, American Society of Mechanical Engineers.

6. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

7. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Asscciated Components of Water-Cooled Nuc ower Plants."

8. Regulatory Guide 1.38, "Quality Assurance Requirements in Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Plants."

9. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

10. Standard Review Plan 5.2.3, "RCPB Materials."

11. Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing."

12. Standard Review Plan 5.3.1, "Reactor Vessel Materials."

13. Standard Review Plan 5.3.2, "Pressure-Temperature Limits."

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PREFACE TO SECTION 5.4

The Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants, Revision 2, under 5.4, "Component and Subsystem Design," contains several paragraphs that provide examples of principal components and subsystems within or allied with the reactor coolant system. The technical review of many of these principal items is conducted under the cognizance of Standard Review Plans which have been prepared for other sections of the Standard Format.

An outline of the material to be reviewed and the Standard Review Plan (SRP) under which it is covered is presented below for those items reviewed in other plans.

5.4.1 Reactor Coolant Pumps

The RSB under SRP 4.4 reviews the process design parameters. Flow coastdown and startup characteristics are reviewed under the Chapter 15 SRP.

The MEB under SRP 3.9.1-3.9.3 and 3.9.6 reviews the structural integrity and operability, the methods of analysis, and the dynamic testing of pumps.

The MTEB under SRP 5.2.4 reviews the inservice inspection and functional testing and under SRP 5.2.3 the materials of fabrication.

5.4.2 Steam Generators (PWR)

The RSB under SRP 4.4 reviews the configuration and process design parameters. The response to various anticipated transients and accidents is reviewed under various SRP for Chapter 15.

The MEB under SRP 3.9.1-3.9.3 reviews the structural adequacy, the methods of analysis, and the dynamic testing of the steam generator.

The MTEB reviews the materials and the inservice inspection program under SRP 5.4.2.1 and 5.4.2.2.

5.4.3 Reactor Coolant Piping

The RSB under SRP 4.4 reviews the piping and instrumentation diagrams, process flow features, and equipment arrangements.

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The MEB under SRP 3.9.1-3.9.3 reviews the structural integrity, the methods of analysis, and the dynamic testing of the reactor coolant piping.

The MTEB reviews the materials and inservice inspection program under SRP 5.2.3 and 5.2.4.

5.4.4 Main Steam Line Flow Restrictions

The RSB reviews the functional requirements under the Chapter 15 SRP.

The MEB reviews the mechanical design bases, methods of analysis, and dynamic testing under SRP 3.9.1-3.9.3.

5.4.5 Main Steam Line Isolation System

This is covered in SRP 9.5.9. The APCSB has primary review responsibility.

5.4.6 Reactor Core Isolation Cooling System (BWR)

A review plan is provided on this subject. The RSB has primary responsibility.

5.4.7 Residual Heat Removal (RHR) System

A review plan is provided on this subject. The RSB has primary responsibility.

5.4.8 Reactor Coolant Cleanup System (BWR)

A review plan is provided on this subject. The ETSB has primary responsibility.

5.4.9 Main Steam Line and Feedwater Piping

The APCSB reviews the functional and related requirements under SRP 10.3 and 10.4.9.

The MEB reviews the structural integrity, methods of analysis, and dynamic testing under SRP 3.9.1-3.9.3.

The MTEB reviews the materials of fabrication under SRP 10.3.6.

5.4.10 Pressurizer

The RSB reviews the configuration and process design parameters under SRP 4.4. Related safety and relief valve capacities are reviewed under SRP 5.2.2. Performance under system transients is reviewed under appropriate SRP of Chapter 15.

The MEB reviews the structural integrity, methods of analysis, and dynamic testing under SRP 3.9.1-3.9.3.

The MTEB reviews the inservice testing and inspection under SRP 5.2.4, and the materials under SRP 5.2.3.

5.4.11 Pressurizer Relief Tank

A review plan is provided on this subject. The APCSB has primary responsibility.

5.4.12 Valves

The RSB reviews the functional aspects of valves within and connected to the reactor coolant pressure boundary under the cognizant review plan. These review plans include SRP 5.4.6, 5.4.7, and 6.3.

The MEB reviews the structural integrity and operability, the methods of analysis, the dynamic testing, and functional testing of valves under SRP 3.9.1-3.9.3 and 3.9.6.

The MTEB reviews valve materials under SRP 5.2.3, 6.1.1, and 10.3.6, and inservice inspection under SRP 5.2.4 and 6.6.

5.5.13 Safety and Relief Valves

The RSB reviews set points and capacities of primary coolant system safety and relief valves under SRP 5.2.2.

The MEB reviews the structural integrity, methods of analysis, dynamic testing, and functional testing under SRP 3.9.1-3.9.3 and 3.9.6.

The MTEB reviews the inservice inspection program under SRP 5.2.4 and 6.6, and materials under SRP 5.2.3, 6.1.1, and 10.3.6.

5.4.14 Component Supports

The MEB reviews the structural integrity, methods of analysis, and dynamic testing of component supports under SRP 3.9.1-3.9.3.

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SECTION 5.4.1.1 PUMP FLYWHEEL INTEGRITY (PWR)

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

1. AREAS OF REVIEW

General Design Criterion 4 requires that structures, systems, and components of nuclear power plants important to safety be protected against the effects of missiles that might result from equipment failures. Because flywheels have large masses and rotate at speeds of 900 rpm or 1200 rpm during normal reactor operation, a loss of integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

The following areas relating to reactor coolant pump flywheel integrity are reviewed:

1. Materials Selection

Reactor coolant pump flywheels are of a simple geometric shape, and are made of ductile material. Their quality can be closely controlled and their service conditions are not severe; therefore, the use of suitable material, coupled with adequate design and inservice inspection can provide a sufficiently small probability of a flywheel failure that the consequences of failure need not be protected against.

Information in the applicant's safety analysis report (SAR) on materials selection and the procedures used to minimize flaws and improve mechanical properties is reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the flywheel materials.

2. Fracture Toughness

The fracture toughness of the materials, including materials tests, correlation of Charpy specimens to fracture toughness parameters, or the alternate use of a nil-ductility transition reference temperature (RT_{NDT}), are reviewed to establish that the flywheel materials will exhibit adequate fracture toughness at normal operating temperature.

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3. Preservice Inspection

The descriptive information is reviewed to verify that the bore of the flywheel is machined to final dimensions if it is flame cut, and that ultrasonic and surface inspections are performed on all finished machined surfaces.

4. Flywheel Design

The flywheel design information including allowable stresses, design overspeed considerations, and shaft and bearing design adequacy, is reviewed.

5. Overspeed Test

The applicant's overspeed test procedures are reviewed to establish their adequacy.

6. Inservice Inspection

A description of the preservice and postoperational phases of the inservice inspection program, including types of inspections, areas inspected, frequencies of inspection, and flaw acceptance criteria, is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are provided in Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," and are as follows:

1. Materials Selection

The applicant's selection of flywheel material is acceptable if it is in accordance with the following criteria:

The flywheel material must be produced by a process (such as vacuum melting or degassing) that minimizes flaws in the material and improves its fracture toughness properties. The material must be examined and tested to meet the following criteria:

a. The nil-ductility transition (NDT) temperature of the flywheel material, as obtained from drop-weight tests (DWT) performed in accordance with the specification ASTM E-208 (Ref. 3), should be no higher than 10°F.

b. The Charpy V-notch (C_v) upper-shelf energy level in the "weak" direction (WR orientation in plates) of the flywheel material should be at least 50 ft-lbs. A minimum of three C_v specimens should be tested from each plate or forging, in accordance with ASTM A-370 (Ref. 4).

2. Fracture Toughness

The following fracture toughness criteria are derived from Regulatory Guide 1.14, C.1.c, and the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Appendix G. The pump flywheel fracture toughness properties are acceptable if they are in compliance with the following criteria:

The minimum static fracture toughness of the material at the normal operating temperature of the flywheel should be equivalent to a critical stress intensity factor, K_{Ic}, of at least 150 ksi $\sqrt{\text{in.}}$ Compliance can be demonstrated by either of the following:

a. Testing of the actual material to establish the K_{Ic} value at the normal operating temperature.

b. Determining that the normal operating temperature is at least 100°F above the RT_{NDT}.

3. Preservice Inspection

The following preservice inspection criteria are derived from Regulatory Guide 1.14, C.1.d, C.1.e, and C.1.f. The applicant's preservice inspection program including finish machining and ultrasonic and surface inspections is acceptable if in compliance with the following criteria:

a. Each finished flywheel should be subjected to a 100% volumetric examination by ultrasonic methods using procedures and acceptance criteria specified in Code Section III, NB-2530 for plates, and NB-2540 for forgings.

b. If the flywheel is flame cut from a plate or forging, at least 1/2 inch of material should be left on the outer and bore radii for machining to final dimensions.

c. Finish machined bores, keyways, splines, and drilled holes should be subjected to magnetic particle or liquid penetrant examination.

4. Flywheel Design

The following flywheel design criteria are derived from Regulatory Guide 1.14, C.2. The applicant's flywheel design is acceptable if in compliance with the following criteria:

The flywheel should be designed to withstand normal conditions, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake without loss of structural integrity.

The design of the pump flywheel should meet the following criteria:

a. The combined stresses at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, should not exceed 1/3 of the minimum specified yield strength or 1/3 of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel.

b. The design overspeed of a flywheel should be at least 10% above the highest anticipated overspeed. The anticipated overspeed should include consideration of the maximum rotational speed of the flywheel if a break occurs in the reactor coolant piping in either the suction or discharge side of the pump. The basis for the assumed design overspeed should be submitted to the staff for review.

c. The combined stresses at the design overspeed, due to centrifugal forces and the interference fit, should not exceed 2/3 of the minimum specified yield strength, or 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel.

d. The shaft and the bearings supporting the flywheel should be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake.

5. Overspeed Test

The following overspeed test criterion is taken from Regulatory Guide 1.14, C.3. The applicant's commitment to perform an overspeed test is acceptable if each flywheel assembly is to be tested at the design overspeed of the flywheel.

6. Inservice Inspection (ISI)

The following inservice inspection program criteria are derived from Regulatory Guide 1.14, C.4. The applicant's ISI program is acceptable if in compliance with the following:

a. A volumetric examination by ultrasonic methods of the areas of higher stress concentration at the bore and keyway at approximately 3-1/3 year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by the Code, Section XI. Removal of the flywheel is not required.

b. A surface examination by liquid penetrant or magnetic particle methods of all exposed surfaces, and 100% volumetric examination by ultrasonic methods at approximately ten-year intervals, during the plant shutdown coinciding with the inservice inspection schedule as required by the Code, Section XI. Removal of the flywheel is not required.

c. A preservice baseline inspection incorporating all the procedures of a and b above, which should establish initial flywheel conditions, accessibility, and practicality of the program.

d. Examination procedures and acceptance criteria should be in conformance with the requirements specified in II.3.a.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review, the following review procedure is followed:

1. Materials Selection

The materials selection, including the procedures to minimize flaws and improve mechanical properties described by the applicant, are reviewed and compared with the requirements of Section II.1 of this plan. If it is a new material not used in prior licensing cases, the materials selection is reviewed and evaluated to establish its acceptability. Based on past evaluations, the following materials are suitable for pump flywheels provided that they meet all the criteria listed in II. 1 and II. 2 of this plan: ASME SA-533-B Class 1, ASME SA-508 Class 2, and ASME SA-516 Grade 65 (Ref. 2).

2. Fracture Toughness

The fracture toughness properties of the flywheel materials, including test data where applicable, are reviewed and compared with the requirements of Section II.2 of this plan. Two alternative methods for deriving the fracture toughness of the flywheel materials are acceptable. The value of the critical stress intensity factor is based on

fracture mechanics testing, while the use of the reference temperature approach is based on the stated normal operating temperature of the flywheel and the actual reference nil-ductility transition temperature of the materials, if an operating license review, or as specified, if a construction permit review.

3. Preservice Inspection

The preservice inspection program, including finish machining, and ultrasonic and surface inspections described by the applicant is reviewed and compared with the requirements of Section II.3 of this plan. The extent to which the ultrasonic inspections proposed and the acceptance criteria in the SAR agree with Code Section III, NB-2530 for plate materials or NB-2540 for forgings, are reviewed.

4. Flywheel Design

The design and stress analysis procedures used for the flywheel are reviewed, including the following areas:

a. Load combinations at normal operating speed and allowable stresses.
b. Design overspeed and basis for selection of design overspeed.
c. Load combinations or design overspeed and allowable stresses.
d. Shaft and bearing load combinations.

The information given in the SAR is compared and evaluated against Section II.4 of this plan.

5. Overspeed Test

The applicant should confirm that an overspeed test will be run in compliance with Section II.5 of this plan.

6. Inservice Inspection

The inservice inspection program described by the applicant in the plant technical specifications, including areas to be inspected, methods of inspection, frequency of inspection, and acceptance criteria, is reviewed and compared with the requirements of Section II.6 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, preservice spin testing, and inservice inspection.

"The applicant's selection of materials, fracture toughness tests, design procedures, preservice overspeed spin testing program, and inservice inspection program for reactor coolant pump flywheels have been reviewed and found acceptable on the basis of conformance with Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," and established industry codes and standards.

"The use of suitable materials with adequate fracture toughness, conservative design procedures, preservice testing, and inservice inspection for flywheels of reactor coolant pump motors provide reasonable assurance of the structural integrity of the flywheels in the event of design overspeed transients or postulated accidents. Conformance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying in part the requirements of General Design Criterion 4."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. ASME Boiler and Pressure Vessel Code, Sections II, III, and XI, American Society of Mechanical Engineers.

3. ASTM E-208-69, "Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

4. ASTM A-370-72, "Methods and Definitions for Mechanicai Testing of Steel Products," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

5. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity" (originally Safety Guide 14).

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SECTION 5.4.2.1 STEAM GENERATOR MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 14, 15, and 31 of Appendix A of 10 CFR Part 50 require that the reactor coolant pressure boundary (RCPB) must have an extremely low probability of abnormal leakage and must be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation and anticipated operational occurrences, and that the probability of rapidly propagating failure of the RCPB is minimized.

A review is made of the following areas reported in the applicant's safety analysis report (SAR). These are all related to the ASME Boiler and Pressure Vessel Code (hereafter "the Code") Class 1 and Class 2 components of pressurized water reactor (PWR) steam generators, including all components that constitute part of the reactor coolant pressure boundary.

1. Selection and Fabrication of Materials

The materials selected for the steam generator are reviewed.

Components of the steam generator are divided into two classes: Class 1, which includes material for those parts exposed to the primary reactor coolant, and Class 2, which includes materials for parts exposed to the secondary coolant water.

The selection and fabrication of materials for all Class 1 and Class 2 components of pressurized water reactor (PWR) steam generators is reviewed for adequacy and suitability and for compliance with the requirements of the Code.

Examples of materials that are currently being used for Class 1 components include the following:

Tubing	-	ASME SB-163, Ni-Cr-Fe, annealed (Inconel 600)
Tube Sheet	-	ASME SA-533, Grade A, weld-Clad with Inconel 600 on the primary coolant side
Channel Head Casting or	-	ASME SA-216, Grade WCC, Class 1, weld-clad with austenitic stainless steel

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Channel Head Plate	-	ASME SA-533, Grade A, B, or C
Forged Nozzles	-	ASME SA-503, Class 2

Examples of materials that are currently being used for Class 2 components include the following:

Shell Pressure Plates	-	ASME SA-533, Grade A, B, or C, Class 2
Bolting	-	ASME SA-193, Grade B-7
		ASME SA-540, Grade B 23 or B 24
Tube Support Plates or Grids	-	ASME SA-240
		ASME SA-479

The corrosion adequacy and suitability of all materials are reviewed. The fracture toughness properties and requirements for ferritic materials of Class 1 and Class 2 components are reviewed.

2. Steam Generator Design

The design and the fabrication procedures are reviewed to determine that the extent of crevice areas are minimized in the completed steam generators. A "tube denting" phenomenon has occurred in a number of steam generators. Based on operating experience and laboratory testing, it is believed that the denting is associated with the growth of a corrosion product (principally Fe_3O_4) in the crevice. The corrosion is caused by the concentration of steam generator water impurities in the annulus. The growth of corrosion product puts inward pressure on the tube resulting in radial deformation of the tube. As corrosion proceeds and in-plate forces accumulate, there are a number of secondary effects in the steam generator. These include (a) tube support plate hole dilation; (b) tube support plate flow hole distortion, flow slot hour-glassing; (c) tube support plate expansion with cracking between hole ligaments; (d) wrapper distortion; (e) leg displacement of the smallest radius U-bend heat tube, and (f) tube leakage.

The extent of the tube to tube sheet contact and the contact area of the tube/tube support are of particular interest. The reviewer will evaluate the design and material selection used to minimize the support plate corrosion.

The tubes are commonly welded to the tube-sheet cladding and expanded into the tube sheet by rolling or explosive-expanding (explanding). Full depth expansion is the preferred design.

3. Compatibility of the Steam Generator Components with the Primary and Secondary Coolant

The possibility of stress-corrosion cracking, denting, pitting, and wastage of the tubes as determined by the chemistry of both the primary and secondary coolants, are reviewed. The methods to be used in monitoring and maintaining the chemistry of the secondary coolant within the specified ranges are reviewed. The compatibility of austenitic and ferritic stainless steels, ferritic low alloy steels and carbon steels with the primary and secondary coolants is reviewed.

4. Cleanup of Secondary Side

The provisions for access to as well as the procedures and methods for the removal of surface deposits, sludge, and corrosion products from the secondary side of the steam generator are reviewed. These provisions are to supplement the removal of sludge by blowdown.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Selection and Fabrication of Materials

a. The acceptable materials for steam generator components are those identified and permitted in the ASME Code, Appendix I of Section III, and specified in detail in the Code Parts A, B, and C of Section II. Any materials specified in the design to meet code-case requirements must also meet the requirements given in Regulatory Guide 1.85, "Code Case Acceptability - Materials." Any materials selected for the tube support structure should be justified on the basis of minimizing the denting and corrosion of the tubes.

b. The fracture toughness of ferritic materials used for Class 1 components in the steam generator must meet the requirements of Subarticle NB-2300, Section III of the Code and Appendix G, Article G-2000 of the Code.

c. The fracture toughness properties of the ferritic materials selected for Class 2 components in the steam generator must meet the requirements of Subarticle NC-2300 of the Code.

d. Welding qualification, weld fabrication processes and inspection during fabrication and assembly of the steam generator must be conducted in conformance with the requirements of Section III and IX of the Code.

e. The corrosion-resistant weld-deposited cladding on the tube sheet and on other primary side components must be fabricated and inspected according to the requirements given in Articles I, II, III, and IV, Part QW of Section IX of the Code.

f. The welds between the tubes and the tube sheet must meet the requirements of Section III and Section IX of the Code.

g. Onsite cleaning and cleanliness control should be in accordance with the position given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and in ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

h. The processing and heat treatment of the steam generator tubing will be evaluated on a case basis. Special heat treatment to improve the corrosion resistance of the tubing should have supporting data.

2. Steam Generator Design

The steam generators must be designed to avoid extensive crevice areas where the tubes pass through the tube sheet, and where the tubes pass through tubing supports, as indicated in Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators."

At the tube/tube sheet interface, the tubes should be rolled or expanded for the full depth of the tube sheet to avoid the presence of a crevice. The tube support structure should be designed to promote high velocity flow along the tubes. This will minimize the buildup of corrosion product and sludge in the crevices of the tube/tube support structure.

3. Compatibility of the Steam Generator Tubing with the Primary and Secondary Coolant

The acceptance criteria for primary coolant chemistry are given in Standard Review Plan 5.2.3, "RCPB Materials." The secondary coolant purity should be monitored as described in Branch Technical Position MTEB 5-3.

4. Cleanup of Secondary Side

The steam generators must be designed to provide adequate access to the internals so that tools may be inserted to inspect and clean up deposits, on the tube sheet and on the tube/tube support. Procedures, such as lancing to remove deposits, should be described.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review, the following review procedure is followed:

1. Selection and Fabrication of Materials

The reviewer examines the materials and fabrication procedures as given in the SAR for Class 1 and Class 2 components of the steam generators, to determine the degree of conformance with the acceptance criteria stated in Section II.1, and verifies that information relative to toughness tests is in conformance with the acceptance criteria stated in Section II.1. The reviewer verifies that the tubes are properly welded and expanded into the tube sheet, and that proper care is taken to maintain cleanliness during fabrication, assembly, and installation of the unit.

2. Steam Generator Design

The reviewer examines the design of the steam generators to verify that tight crevice areas where tubes pass through the tube supports and tube plate(s) are minimized, as discussed in Section II.2.

3. Compatibility of the Steam Generator Tubing with the Primary and Secondary Coolant

The reviewer examines the controls to be placed on the composition of the primary and secondary coolants to determine that they meet the acceptance criteria cited in Section II.3.

4. Cleanup of Secondary Side

The reviewer examines the design provisions that allow implementation of the procedures and methods to be used for removal of surface deposits, sludge, and corrosion products from the tube sheet and the tube/tube support areas.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided in accordance with the requirements of this review plan and that his evaluation supports conclusions of the following type, which are to be included as applicable in the staff's safety evaluation report:

"The materials selected for use in Class 1 and Class 2 components will be fabricated and inspected in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the steam generator will be down in conformance with the requirements of Section III and IX of the ASME Code.

"The primary side of the steam generator is designed and fabricated to comply with ASME Class 1 criteria as required by the staff. (The secondary side pressure boundary part of the steam generator will be designed, manufactured, and tested to ASME Class 1 criteria although the staff required classification is ASME Class 2.)*

"The pressure boundary materials of ASME Class 1 components of the steam generator will comply with the fracture toughness requirements and tests of Subarticle NB-2300 of Section III of the Code. The materials of the ASME Class 2 components of the steam generator will comply with the fracture toughness requirements of Subarticle NC-2300 of Section III of the Code.

"The crevice between the tube sheet and the inserted tube will be minimal because the tube will be expanded to the full depth of insertion of the tube in the tube sheet. The tube expansion and subsequent positive contact pressure between the tube and the tube sheet will preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling.

"(The tube support plates will be manufactured from ferritic stainless steel material, which has been shown in laboratory tests to be corrosion resistant to the operating environment.)* (The tube support plates will be designed and manufactured with broached holes rather than drilled holes. The broached hole design promotes high

*Include material within parentheses as applicable.

velocity flow along the tube, sweeping impurities away from the support plates locations.) (The tube support structure will be manufactured to the egg crate design. The egg crate design eliminates the narrow annular gap at the tube supports, because the support may contact the tube at only four lines on the tube circumference, and provides almost complete washing of the tube surface with steam generator water.)*

"The onsite cleaning and cleanliness controls during fabrication (will)* conform to the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The controls placed on the secondary coolant chemistry are in agreement with staff technical positions.

"Reasonable assurance of the satisfactory performance of steam generator tubing and other generator materials is provided by (a) the design provisions and the manufacturing requirements of the ASME Code, (b) rigorous secondary water monitoring and control, and (c) the limiting of condenser in-leakage. The controls described above combined with conformance with applicable codes, standards, staff positions, and Regulatory Guides constitute an acceptable basis for meeting in part the requirements of General Design Criteria 14, 15, and 31."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary," Criterion 15, "Reactor Coolant System Design," and Criterion 31, "Fracture Prevention of the Reactor Coolant Pressure Boundary."

2. ASME Boiler and Pressure Vessel Code, Parts A, B, and C of Section II, Section III, and Section IX, American Society of Mechanical Engineers.

3. ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," American National Standards Institute.

4. Regulatory Guide 1.37, "Quality Assurance Requirements of Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.85, "Code Case Applicability-Material."

6. Standard Review Plan 5.2.3, "RCPB Materials."

7. Branch Technical Position - MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," appended.

*Include material within parentheses as applicable.

MONITORING OF SECONDARY SIDE WATER CHEMISTRY IN PWR STEAM GENERATORS

A. BACKGROUND

Effective long-term reliable operation of PWR steam generators requires that operational procedures, design, and selection of materials be such that there is no leakage across the steam generator tubes and that the barrier between the primary and secondary fluids maintains its integrity under any condition as stated in General Design Criteria 14, 15, and 31 of Appendix A of 10 CFR 50.

These objectives are generally met by providing a water treatment to remove impurities from the water, operation procedures to remove impurities from generators, design of equipment to prevent impurities from entering the system, and design factors to prevent the impurities from concentrating and forming sludges or deposits, especially in crevices.

Less than thoroughly effective water treatment, operational procedures, and design factors have led to the degradation of steam generator tubing. An extensive history of stress corrosion cracking, wastage, and denting of steam generator tubing in operating PWRs, has developed; therefore we recommend the following criteria.

B. Branch Technical Position

1. Crevices between the tubing and the tube sheets or tubing supports should be minimized to prevent concentration of impurities or solids in these areas. Steam generators should be designed and built to achieve this goal.

   To minimize the deposition of corrosion products and sludge between the tubes and the supporting structure, the tube/tube support interface should be designed to promote high velocity water flow at the interface. This will improve the "washing" of this area.

2. Condenser cooling water in-leakage to the condensate has been identified as the major source of impurity ingress in the PWR secondary feedwater. The combination of impurity ingress with corrosion of copper containing alloys and corrosion product transport (Fe_3O_4, SiO_2, etc.) in the secondary water system produces sludge that is difficult to remove and is reactive to steam generator materials.

3. The methods utilized for control of secondary side water chemistry should be described. In plants having more than one steam generator, additives to each steam generator should be controlled separately. Records should be made of the following items, and summaries of the data should be available for report as requested by the Commission.

a. For plants utilizing volatile chemistry:

(1) The composition, quantities, and addition rates of additives should be recorded initially and thereafter whenever a change is made.

(2) The electrical conductivity and the pH of the bulk steam generator water and feedwater should be measured continuously.

(3) For once-through steam generators, the pH and electrical conductivity at the coolant inlet should be measured continuously.

(4) Free hydroxide concentration and impurities (particularly chloride, ammonia and silica) in the steam generator water should be measured at least three times per week.

(5) The electrical conductivity of the condensate should be measured continuously. When the conductivity changes, the cation or anion concentration (as applicable to the specific power plant) should be measured to determine if a condenser leak exists.

(6) When a condenser leak is confirmed, the leak should be repaired or plugged within 96 hours.

b. For plants utilizing phosphate treatment:

(1) The composition, quantity, and addition rate of each additive should be recorded initially and thereafter whenever a change is made.

(2) The Na/PO_4 molar ratio of the secondary coolant should be recorded initially and whenever a change is made.

(3) The electrical conductivity and pH of the bulk steam generator water and feedwater should be measured continuously.

(4) The concentration of suspended/dissolved solids and impurities (particularly free caustic, chloride, silica, and sodium) in the steam generator water should be measured daily.

(5) The concentration of dissolved solids (particularly sodium and phosphate) in the blowdown liquid should be measured once each week.

(6) The rate of blowdown should be recorded initially and whenever a change in rate is made.

(7) The hideout and reverse hideout of phosphate should be recorded. The phosphate concentration in each steam generator (or in one steam generator if this is shown to be representative of all) and in the blowdown liquid should be measured before and after each planned power level change of 10% or greater, and should be measured after each unplanned power level change of 20% or greater.

(8) The electrical conductivity of the condensate should be measured continuously. When the conductivity changes, the cation or anion concentration (as applicable to the specific power plant) should be measured to determine if a condenser leak exists.

(9) When a condenser leak is confirmed, the leak should be repaired within 96 hours.

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SECTION 5.4.2.2 STEAM GENERATOR INSERVICE INSPECTION

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 1 and 32 of Appendix A of 10 CFR Part 50 require that components which are part of the reactor coolant pressure boundary (RCPB) or other components important to safety be designed to permit periodic inspection and testing of critical areas for structural and leaktight integrity. The design of the steam generators as described in the preliminary safety analysis report (PSAR) is reviewed to establish that use of the specified inspection techniques is feasible. The provisions made for baseline inspection prior to startup, the methods to be used for the inspections, and the inservice inspection program are reviewed in the final safety analysis report (FSAR) and plant Technical Specifications.

II. ACCEPTANCE CRITERIA

The design of the steam generators to provide access for an inservice inspection (ISI) program, and the proposed ISI program should follow the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section XI, and the recommendations given in Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes." Specifically, the steam generators should be designed to permit inspection of any component, including individual tubes, and the repair or replacement of any component. The tube examination equipment and procedures should be capable of detecting and locating defects with a penetration of 20% or more of the wall thickness. A permanent record of test data should be provided. A baseline tube inspection should be scheduled prior to startup. The sample selection and testing of tubes, the inspection intervals, and the actions to be taken if defects are identified should follow the recommendations of Regulatory Guide 1.83.

Section XI provides for the volumetric inspection of the following components: longitudinal and circumferential welds, including tube-sheet-to-head or shell welds on the primary side; primary nozzle-to-vessel head welds and nozzle-to-head inside radiused section; primary nozzle to safe-end welds; primary side pressure-retaining bolting; integrally-welded primary vessel supports; circumferential butt welds on the secondary shell side; and

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nozzle-to-vessel welds on the secondary side. The volumetric inspections are supplemented, as detailed in the Code, by visual and surface inspections for both primary and secondary steam generator Code Class 1 and 2 components.

III. REVIEW PROCEDURES

The reviewer will select and .mphasize material from the procedures described below, as may be appropriate for a particular case. He determines that the design of the steam generators, as described in the PSAR, will permit access for the specified inspection techniques of all steam generator components including tubes. He also evaluates the design of the steam generator as described in the FSAR and the Technical Specification inservice inspection program to determine the degree to which the recommendations of Regulatory Guide 1.83 and Code Section XI have been followed. The reviewer determines that the techniques to be used for inservice inspection are those listed in Section XI. He determines that the inspection technique for the tubes, the selected number of tube samples, the inspection intervals, and the actions to be taken in the event defects are observed are in accordance with the positions stated in the guide. He determines that a baseline inspection will be made prior to startup of the plant. Standard Review Plan 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," covers the ISI of Class 1 components of the steam generators. Standard Review Plan 6.6, "Inservice Inspection of Class 2 and 3 Components," is applicable to ISI of the balance of the components of the steam generator, which are all Class 2 components.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The steam generators have been designed to permit inservice inspection of all Code Class 1 and 2 components including individual tubes as recommended in Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes," and ASME Code Section XI. [The inservice inspection program for the steam generators is in accordance with the recommendations of the above-cited Regulatory Guide and ASME Code Section XI with respect to the inspection methods to be used, provisions for a baseline inspection, selection and sampling of tubes, inspection interval, and actions to be taken in the event defects are identified.*] Conformance with Regulatory Guide 1.83 and ASME Code Section XI constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 32."

V. REFERENCES

1. 10 CFR 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

*The statement enclosed in brackets should only be included in the staff's safety evaluation report for an FSAR.

3. ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

4. Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes."

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SECTION 5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Instrumentation and Control Systems Branch (ICSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The reactor core isolation cooling (RCIC) system in a boiling water reactor (BWR) is a safety system which serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. Abnormal events which could cause such a situation to arise include an inadvertent isolation of all main steam lines, loss of condenser vacuum, pressure regulator failure, loss of feedwater, the loss of offsite power, and total loss of all a-c power (both offsite and diesel generators). Each of these transients is analyzed in Chapter 15 of the applicant's safety analysis report (SAR). For each of these events, the high pressure part of the emergency core cooling system (ECCS) provides a backup function to the RCIC system. This review of the RCIC is performed to assure conformance with the requirements of General Design Criteria 2, 4, 34, 55, 56 and 57.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel and supplying steam to and removing condensate from the RCIC steam turbine where applicable. Fluid removed from the reactor vessel following a shutdown from power operation is normally made up by the feedwater system, supplemented by in-leakage from the control rod drive system. If the feedwater system is inoperable, the RCIC turbine-pump unit starts automatically or is started by the operator from the control room. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool.

The review of the RCIC system includes the system design bases, design criteria, description, and the points noted below.

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1. The piping and instrumentation diagram are reviewed to determine that the system is capable of performing its intended function and of being preoperationally and operationally tested.

2. The degree of separation from the high pressure core spray (HPCS) system, or high pressure core injection (HPCI) system for 1967 product line or earlier BWR's, and protection against common mode failures of both redundant systems (e.g., from flooding, fire, pipe whip, or high temperature, pressure, and humidity) are reviewed.

3. The process flow diagram is reviewed to confirm that the RCIC system design parameters are consistent with expected pressures, temperatures, and flow rates.

4. The complete sequence of operation is reviewed to determine that the system can function as intended and that the system is capable of manual operation.

5. The proposed preoperational and initial startup test programs are reviewed to determine their adequacy.

6. The proposed technical specifications are evaluated to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

7. The RCIC system is reviewed to assure that it has the proper seismic and quality group classifications. This aspect of the review is performed as part of the effort described in SRP Sections 3.2.1 and 3.2.2. The RCIC system is to be enclosed in a structure having the proper seismic classification. The review of the building seismic category is also accomplished as a portion of the effort described in SRP Section 3.2.2.

   The RCIC is to be enclosed in a seismic Category I structure or building. The design adequacy of this structure or building is evaluated by SEB as described in SRP Sections 3.3, 3.4, 3.5, 3.7 and 3.8.

8. The CSB reviews the RCIC system, as described in SRP Section 6.2.4, to confirm that the design is compatible with the containment system and can be isolated.

9. The ICSB, as described in SRP Section 7.4, evaluates the adequacy of controls and instrumentation of the RCIC system with regard to the required features of automatic actuation, remote sensing and indication, and remote control.

10. The PSB, as described in SRP Section 8.3, evaluates the adequacy of emergency onsite power, sufficient battery capacity, and use of d-c power only.

11. The MEB, as described in SRP Section 3.9.3, ensures that the design and installation of the RCIC system meet applicable codes and are adequate for its proper functioning.

12. The CPB, on request, reviews the core decay energy output on which the design is based to see that it is applicable and suitably conservative.

13. The MTEB reviews the materials and the inservice inspection program for the RCIC system.

II. ACCEPTANCE CRITERIA

The general objective of the review is to determine that the RCIC system, in conjunction with the HPCS (or HPCI) system, meets the requirements of General Design Criteria 34 (Ref. 3) by providing the capability for decay heat removal to allow complete shutdown of the reactor under conditions requiring its use. It must maintain the reactor water inventory until the reactor is depressurized sufficiently to permit operation of the low pressure cooling systems. The RCIC system, in conjunction with the HPCS (or HPCI) system, must be capable of removing fission product decay heat and other residual heat from the reactor core following shutdown so as to preclude fuel damage or reactor coolant pressure boundary overpressurization. For the purposes of this SRP section the minimum critical power ratio (MCPR) should be greater than X*, based on Reference 7, to preclude fuel damage. The maximum reactor pressure should be less than 110% of design pressure (Ref. 8).

Historically, credit has been taken for RCIC system capability to mitigate the consequences of certain abnormal events; however, since the cooling function is redundant to the HPCI or HPCS system, the RCIC system itself is not required to meet the single failure criterion. However, the system is to perform its function without the availability of any a-c power. As a system which must respond to certain abnormal events, the RCIC system must be designed to seismic Category I standards, as defined in Regulatory Guide 1.29 (Ref. 9).

The RCIC and HPCS (or HPCI) systems must be protected against natural phenomena, external or internal missiles, pipe whip, and jet impingement forces so that such events cannot fail both systems simultaneously. Jointly, the two systems must meet General Design Criterion 2 (Ref. 1); General Design Criterion 4 (Ref.2); the recommendations of Regulatory Guide 1.46 (Ref. 10); and the staff positions on protection for pipe failures outside containment (Ref. 12).

The RCIC system must meet the requirements of General Design Criteria 55, 56, and 57 (Refs. 4, 5, and 6) with regard to isolation provisions for lines passing through the primary containment.

If the RCIC system is used to control or mitigate the consequences of an accident, either by itself or as a backup to another system, it must meet the requirements of an engineered safety feature.

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*The value of MCPR will vary for different product lines. The value of MCPR used for a particular case review is to be consistent with the value specified in the plant technical specifications as the fuel integrity safety limit.

The preoperational and initial startup test programs for the RCIC system should meet the intent of Regulatory Guide 1.68 (Ref. 11).

For those areas of review identified in Subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Subsection II.

For the operating license (OL) review, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The following steps are taken by the reviewer to determine that the acceptance criteria of Subsection II have been met. The steps are adapted to CP or OL reviews as appropriate.

1. Using the RCIC operating requirements specified in SAR Section 5.4.6 and Chapter 15, the reviewer confirms that the RCIC can function when required so as to prevent the critical power ratio from decreasing below X* (based on Reference 7) and prevent the reactor pressure from exceeding 110% of design pressure. This determination is based on engineering judgment and independent calculations (where deemed necessary), using information as specified in steps 2 and 3 below. The reviewer consults with the CPB to assure that the decay heat loads used in the RCIC analyses are applicable and suitably conservative. The reviewer also determines that the RCIC system maintains sufficient coolant inventory in the reactor vessel to keep the core covered and assure clad integrity.

2. Using the description given in Section 5.4.6 of the SAR, including component lists and performance specifications, the reviewer determines that the RCIC system piping and instrumentation are such as to allow the system to operate as intended. This is accomplished by reviewing the piping and instrumentation diagrams to confirm that piping arrangements permit the required flow paths to be achieved and that sufficient process sensors are available to measure and transmit required information.

3. Using the comparison tables of SAR Section 1.3, the RCIC system is compared to designs and capacities of such systems in similar plants to see that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

*IBID. Page 5.4.6-3

4. The reviewer checks the piping and instrumentation diagrams and equipment layout drawings for the RCIC and HPCS (or HPCI) systems to see that the systems are physically separated and can function independently, and that they jointly conform to the requirements of General Design Criteria 2 and 4 and the recommendations of Regulatory Guide 1.46 and staff positions on piping failures outside containment (Refs. 1, 2, 10, and 12).

5. Based on the description in SAR Section 5.4.6, the reviewer judges whether adequate control and monitoring information is available to allow the operator to actuate the system manually or to realign the RCIC system manually within the time allowed (i.e., change the RCIC system suction from the condensate storage tank to the suppression pool, if automatic switchover has not been provided, or to the steam condensing mode of the residual heat removal system).

6. The reviewer contacts ICSB to confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. The reviewer contacts PSB to ascertain that the RCIC system operation is not dependent on a-c power sources, and that there is sufficient battery capability to permit operation of the RCIC for a period of two hours without the availability of a-c power. The instrumentation and controls of the RCIC system, in conjunction with the HPCS (or HPCI) system, are to have sufficient redundancy to satisfy the single failure criterion.

7. The reviewer checks the piping and instrumentation diagrams to see that essential RCIC system components are designated seismic Category I.

8. The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 11). At the OL stage, the reviewer assures that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see par. C.2.b of Regulatory Guide 1.68).

   The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the RCIC system will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the RCIC system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the RCIC system. If the RCIC system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the CP stage. This effect should particularly evaluate the need for any special design features required to perform acceptable test programs.

9. The proposed plant technical specifications are reviewed to:

a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

b. Verify that the frequency and scope of periodic surveillance testing is adequate.

10. The reviewer confirms that the RCIC is housed in a structure whose design and design criteria have been reviewed by other branches to assure that it provides adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

11. Upon request from the primary reviewer, other secondary review branches will provide input for the areas of review stated in Subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

"The reactor core isolation cooling (RCIC) system includes the piping, valves, pumps, turbines, instrumentation, and controls used to maintain water inventory in the reactor vessel whenever it is isolated from the main feedwater system. Certain engineered safety features (HPCS or HPCI) provide a redundant backup for this function. The scope of review of the RCIC system for the _____ plant included piping and instrumentation diagrams, equipment layout drawings, and functional specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RCIC system, his analysis of the adequacy of the criteria and bases, and the conformance of the design to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses have been reviewed and have been found to conform to Commission regulations as set forth in General Design Criteria 2, 4, 34, 55, 56, and 57, and to applicable regulatory guides and staff technical positions. The RCIC system has been found to conform to the recommendation of Regulatory Guide 1.29. The RCIC system and HPCS (or HPCI) system jointly conform to General Design Criteria 2, 4, 34, and the recommendations of Regulatory Guide 1.46. The two systems have been found capable of removing core decay heat following feedwater system isolation and reactor shutdown so that the critical power ratio does not decrease below _____, and the pressure within the reactor coolant pressure boundary does not exceed 110% of design pressure. This capability has been found to be available even with a loss of offsite power and with a single active failure. The staff concludes that the design of the reactor core isolation cooling system conforms to the Commission's regulations and to applicable regulatory guides and staff positions, and is acceptable.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

4. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

5. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

6. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

7. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Applicable," NEDO-10958, General Electric Company, November 1973.

8. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

9. Regulatory Guide 1.29, "Seismic Design Classification."

10. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

11. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

12. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.4.7 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Auxiliary Systems Branch (ASB)
Core Performance Branch (CPB)
Containment Systems Branch (CSB)
Instrumentation and Control Systems Branch (ICSB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The residual heat removal (RHR) system is used in conjunction with the main steam and feedwater systems (main condenser), or the reactor core isolation cooling (RCIC) system in conjunction with the safety/relief valves in a boiling water reactor (BWR), or auxiliary feedwater system in conjunction with the atmospheric dump valves in a pressurized water reactor (PWR) to cool down the reactor coolant system following shutdown. Parts of the RHR system also act to provide low pressure emergency core cooling and are reviewed as described in SRP Section 6.3. Some parts of the RHR system also provide containment heat removal capability and are reviewed as described in SRP Section 6.2.2. The review by RSB is to ensure that the design of the RHR system is in conformance with General Design Criteria 2, 4, 5, 34, 44, 45, 46, 55, 56 and 57.

Both PWRs and BWRs have RHR systems which provide long-term cooling once the reactor coolant temperature has been decreased by the main condenser, RCIC, or auxiliary feedwater systems. In both types of plants, the RHR is typically a low pressure system which takes over the shutdown cooling function when the reactor coolant system (RCS) temperature is reduced to about 300°F. Although the RHR system function is similar for the two types of plants, the system designs are different.

The RHR system in PWRs takes water from the RCS hot legs, cools it, and pumps it back to the cold legs or core flooding tank nozzles. The suction and discharge lines for the RHR pumps have appropriate valving to assure that the low pressure RHR system is always isolated from the RCS when the reactor coolant pressure is greater than the RHR system design pressure. The heat removed in the heat exchangers is transported to the ultimate heat sink by the component cooling water or service water system. In PWRs, the RHR system is also used to fill, drain, and remove heat from the refueling canal during refueling operations; to circulate coolant through the core during plant startup prior to RCS pump operation, and in some to provide an auxiliary pressurizer spray.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

The RHR system in BWRs is typically composed of four subsystems. The containment heat removal and low pressure emergency core cooling subsystems are discussed in SRP Sections 6.2.2 and 6.3. The shutdown cooling and steam condensing (via RCIC) subsystems are covered by this SRP section. These subsystems make use of the same hardware, consisting of pumps, piping, heat exchangers, valves, monitors, and controls. In the shutdown cooling mode, the BWR RHR system can also be used to supplement spent fuel pool cooling. As in the PWR, the low pressure RHR piping is protected from high RCS pressure by isolation valves.

The steam condensing mode of RCIC operation in BWRs (when included in the plant design) provides an alternative to the main condenser or normal RCIC mode of operation during the initial cooldown. Steam from the reactor is transferred to the RHR heat exchangers where it is condensed. The condensate is piped to the suction side of the RCIC pump. The RCIC pump returns the condensate to the reactor vessel. The heat removed in the heat exchangers is transported to the ultimate heat sink by the service water system.

Other means of removing decay heat in the event that the RHR system is inoperable have been proposed for some BWRs. These approaches use some of the piping that is used for the steam condensing mode of RCIC. These approaches are also covered by this SRP section.

The reactor coolant temperatures and pressure must be decreased before the low pressure RHR system can be placed in operation; therefore, the review of the decay heat removal function must consider all conditions from shutdown at normal reactor operating pressure and temperature to the cold depressurized condition. This effort is divided between the RSB and the ASB as follows:

1. For BWRs, the RSB reviews the processes and systems used in the cooldown of the reactor for the entire spectrum of potential reactor coolant system pressures and temperatures during decay heat removal.

2. For PWRs, the RSB reviews the approach used to meet the functional requirements of BTP RSB 5-1 with respect to cooldown to the conditions permitting operation of the RHR system. Since an alternate approach to that normally used for cooldown may be specified, the reviewers identify all components and systems used. The ASB has primary review responsibility for the review of the pertinent portions of the CVCS (SRP 9.3.4), the atmospheric dump valves (SRP 10.3), and the source for auxiliary feedwater (SRP 10.4.9) for conformance to BTP RSB 5-1. The RSB reviews the pressurizer relief valve and ECCS, if used. In addition, the RSB reviews the tests and supporting analysis concerning mixing of borated water and cooldown under natural circulation as required in BTP RSB 5-1.

3. For both PWRs and BWRs, the ASB reviews the component cooling or service water systems that transfer decay heat from the RHR system to the ultimate heat sink as described in SRP Sections 9.2.1 and 9.2.2.

4. The RSB reviews the design and operating characteristics of the RHR system with respect to its shutdown and long-term cooling function. Where the RHR system interfaces with other systems (e.g., RCIC system, component cooling water system) the effect of these systems on the RHR system is reviewed. Overpressure protection provided by the valving between the RCS and RHR system is also reviewed.

The proposed RHR system preoperational and initial startup test programs are reviewed and the proposed technical specifications are evaluated in regard to limiting conditions of operation and periodic surveillance testing.

The RHR system is reviewed to assure that it has the proper seismic and quality group classifications. This aspect of the review is performed as a portion of the effort described in SRP Sections 3.2.1 and 3.2.2. The RHR system is to be enclosed in a structure having the proper seismic classification. The review is done as a part of the effort described in SRP Section 3.2.2.

The RHR system shall be enclosed in a seismic Category I structure or building. The design adequacy of this structure or building is evaluated by SEB as described in SRP Sections 3.3, 3.4, 3.5, 3.7 and 3.8.

The CSB, as described in SRP Section 6.2.4, reviews the design of the RHR system to see that it is compatible with the function of the containment and that adequate isolation capabilities are provided.

The ICSB, as described in SRP Section 7.4, reviews motor-operated valve controls, interlocks, sensors for interlocks, position indicators, and power sources. ICSB determines that the interlocks on motor-operated valves used as barriers between the high and low pressure RHR piping are suitably independent and diverse and that trip signals close the valves when the pressure is too high.

The MEB, as described in SRP Section 3.9.3, reviews the design and installation of the RHR system to see that applicable code requirements are met.

The MTEB reviews the materials and inservice inspection program for the RHR system, as described in SRP Sections 6.1.1 and 6.6.

The CPB reviews the core decay energy output on which the design is based to see that it is applicable and suitably conservative.

The MEB, as described in SRP Section 3.6.2, reviews the effects of pipe breaks both in and outside containment to confirm that requirements for protection of the reactor shutdown systems are met. The ASB, as described in SRP Section 3.6.1, reviews the plant design for protection against piping failures outside containment to assure that such failures would not cause loss of needed safety-related systems and that the plant could be safely shutdown in the event of such failures.

II. ACCEPTANCE CRITERIA

The general objective of the review is to determine that the system or systems employed to remove residual heat meets the requirements of General Design Criterion (GDC) 34 (Ref. 4) regarding residual heat removal, and GDC 19 (Ref. 16) regarding operability from the control room. As noted in subsection I, reactor cooldown covers a wide range of potential reactor coolant temperatures and pressures. A portion of this range is reviewed by the ASB for PWRs. This SRP section deals specifically with the areas of review performed by the RSB. The acceptance criteria are discussed in the following paragraphs.

The system or systems must be capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. The system(s) and the system(s) enclosure must be in conformance with GDC 2 (Ref. 1) and GDC 4 (Ref. 2), and should conform to the recommendations of Regulatory Guide 1.26 (Ref. 17), Regulatory Guide 1.29 (Ref. 12), Regulatory Guide 1.46 (Ref. 13), and the staff positions on protection against piping failures outside containment (Ref. 15).

- The system(s) are to satisfy the functional, isolation, pressure relief, pump protection, and test requirements specified in Branch Technical Position RSB 5-1 (Ref. 11).

Interfaces between the RHR system and RCIC and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to GDC 5 (Ref. 3). Component cooling and service water systems removing heat from the RHR heat exchangers must conform to GDC 44, 45 and 46 (Refs. 5, 6 and 7). Containment isolation provisions for the RHR system must conform to GDC 55, 56 and 57 (Refs. 8, 9 and 10).

In addition to the above criteria, the acceptability of the RHR system may be based on the degree of design similarity with previously approved plants. Deviations from these criteria from other types of RHR systems (e.g., systems that are designed to withstand reactor coolant system operating pressure or systems located entirely inside containment) will be considered on an individual basis.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the Preliminary Safety Analysis Report meet the acceptance criteria given in subsection II.

For operating license (OL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the Final Safety Analysis Report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

As noted in subsections I and II, the RSB review for PWRs is limited to the low pressure - low temperature RHR system. For BWRs, the review is to include all of the systems used to transfer residual heat from the reactor over the entire range of potential reactor coolant temperatures and pressures. The following steps are to be applied by the reviewer for the appropriate systems, depending on whether a PWR or BWR is being reviewed. These steps should be adapted to CP or OL reviews as appropriate.

1. Using the description given in the applicant's Safety Analysis Report (SAR), including component lists and performance specifications, the reviewer determines that the system(s) piping and instrumentation are such as to allow the system(s) to operate as intended, with or without offsite power and given any single active component failure. This is accomplished by reviewing the piping and instrumentation diagrams (P&IDs) to confirm that piping arrangements permit the required flow paths to be achieved and that sufficient process sensors are available to measure and transmit required information. A failure modes and effects analysis (or similar system safety analysis) provided in the SAR is used to determine conformance to the single failure criterion.

2. Using the comparison tables of SAR Section 1.3, the RHR system is compared to designs and capacities of such systems in similar plants to see that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

3. From the system description and P&IDs, the reviewer determines that the isolation requirements of Branch Technical Position RSB 5-1 (Ref. 11) are satisfied.

4. The reviewer determines that the RHR system design has provisions to prevent damage to the RHR pumps in accordance with Branch Technical Position RSB 5-1 (Ref. 11). The reviewer checks the isolation valves in the suction line for potential closure, NPSH requirements, pump runout, and potential loss of miniflow line during pump testing. If operator action is required to protect the pumps, the reviewer evaluates the instrumentation required to alert the operator and the adequacy of the time frame for operator action.

5. Using the system process diagrams, P&IDs, failure modes and effects analysis, and component performance specifications, the reviewer determines that the system(s) has the capacity to bring the reactor to cold shutdown conditions in a reasonable period of time, assuming a single failure of an active component with only either

onsite or offsite electric power available. For the purposes of the review, 36 hours is considered a reasonable time period. The reviewer consults with the CPB to confirm that the proper core decay energy output was assumed for the analysis. The ASB is responsible for the review of the initial cooldown phase for PWRs. Therefore, this review effort is to be coordinated with that branch. For the purposes of the review of both PWRs and BWRs, only the operation of safety grade equipment is to be assumed.

6. The cooldown function is to be reviewed to determine if it can be performed from the control room assuming a single failure of an active component, with only either onsite or offsite electric power available. Any operation required outside of the control room is to be justified by the applicant. Like Item 5, the initial cooldown for PWRs is to be reviewed by ASB.

7. By reviewing the system description and the P&IDs, the reviewer confirms the RHR system satisfies the pressure relief requirements of Branch Technical Position RSB 5-1 (Ref. 11).

8. The reviewer checks the P&IDs to see that essential components of the systems employed to remove residual heat are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III). Based on statements made in SAR Section 5.4.7 or on the reviews made by other branches, the RSB reviewer confirms that the RHR system meets the requirements of GDC 2 (Ref. 1) and GDC 4 (Ref. 2), and conforms to the recommendations of Regulatory Guides 1.29 (Ref. 12) and 1.46 (Ref. 13) and the staff positions on piping failures outside containment.

9. By reviewing the piping arrangement and system description of the RHR system, the reviewer confirms that the RHR system meets the requirements of GDC 5 (Ref. 3) concerning shared systems.

10. The RSB reviewer contacts the ASB reviewer in conjunction with his review of the RHR system heat sink and refueling system interaction to interchange information and assure that the reviews are consistent in regard to the interfacing parameters. For example, the ASB review determines the maximum service or component cooling water temperature. The RSB reviewer then reviews the RHR system description to determine that this maximum temperature has been allowed for in the RHR system design.

11. The RSB reviewer contacts his counterpart in the ICSB to obtain any needed information from their review. Specifically, ICSB confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. The instrumentation and controls of the RHR system are to have sufficient redundancy to satisfy the single failure criterion.

12. The RSB reviewer contacts his counterpart in CSB so that the information needed concerning their reviews will be interchanged.

13. The applicant's proposed initial startup test programs are reviewed to determine that they are consistent with the requirements of Branch Technical Position RSB 5-1 (Ref. 11) and with the intent of Regulatory Guide 1.68 (Ref. 14). At the OL stage, the RSB reviewer assures that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see paragraph C.1.b of Regulatory Guide 1.68).

    The RSB reviewer evaluates the proposed initial test programs to determine if they provide reasonable assurance that the RHR system will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the RHR system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the RHR system. If the RHR system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required initial startup testing programs are reviewed at the CP stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

14. The proposed plant technical specifications are reviewed to:

    a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

    b. Verify that the frequency and scope of periodic surveillance testing is adequate.

15. The reviewer contacts the SEB reviewer to confirm that the systems employed to remove residual heat are housed in a structure whose design and design criteria provide adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

16. For PWRs, the reviewer confirms that the auxiliary feedwater supply satisfies the requirements of Branch Technical Position RSB 5-1 (Ref. 11).

17. The RSB reviewer provides information to other branches in those areas where the RSB has a review responsibility that is not explicitly covered in steps 1-16 above. These additional areas of review responsibility include:

    a. Identification of engineered safety features (ESF) and safe shutdown electrical loads, and verification that the minimum time intervals for the connection of the ESF to the standby power systems are satisfactory.

b. Identification of vital auxiliary systems associated with the RHR system and determination of cooling load functional requirements and minimum time intervals.

c. Identification of essential components associated with the main steam supply and the auxiliary feedwater system that are required to operate during and following shutdown.

18. Upon request from the primary reviewer, other secondary review branches will provide input for the areas of review stated in subsection I of this SRP section. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's Safety Evaluation.

For PWRs

"The residual heat removal function is accomplished in two phases: the initial cooldown phase and the residual heat removal (RHR) system operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system and the atmospheric dump valves. This equipment is used to reduce the reactor coolant system temperature and pressure to values that permit operation of the RHR system. The review of the initial cooldown phase is discussed in Section ___ of the SER. The review of the RHR system operational phase is discussed below. The residual heat removal (RHR) system removes core decay heat and provides long-term core cooling following the initial phase of reactor cooldown. The scope of review of the RHR system for the _____ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHR system and his analysis of the adequacy of those criteria and bases and the conformance of the design to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses associated with the RHR system have been reviewed and have been found to conform to Commission regulations, regulatory guides, and staff technical positions. The RHR system has been found to conform to General Design Criteria 2, 4, 5, 19, 34, 44, 45, 46, 55, 56 and 57, to Regulatory Guides 1.26, 1.29, 1.46 and 1.68, and to Branch Technical Position RSB 5-1. The system was found capable of performing its shutdown cooling functions from the control room with only onsite electrical power or offsite power available, assuming the most restrictive single active component failure. It was also found that two independent and redundant barriers are always

in place between the reactor coolant systems (RCS) and RHR system whenever the RCS pressure is higher than the RHR system design pressure.

"The staff concludes that the design of the residual heat removal system conforms to the Commission's regulations, and to applicable regulations, guides and staff positions, and is acceptable."

For BWRs

"The residual heat removal function is accomplished in two phases: the initial cooldown phase and a low pressure-temperature operation phase. In the event of loss of offsite electrical power, the initial cooldown phase is accomplished using the reactor core isolation cooling (RCIC) system and the safety/relief valves. The low pressure-temperature mode of operation is usually accomplished by the residual heat removal (RHR) system. However, certain single failures can render the RHR system inoperative. In that event, two alternate systems that use components of the RCIC and RHR system are available to bring the reactor to cold shutdown conditions.

"The scope of review of these systems for the _____ plant included piping and instrumentation diagrams, equipment layout drawings, failure mode and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for these systems and his analysis of the adequacy of those criteria and bases and of the conformance of the design to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses associated with the systems employed to remove residual heat from the reactor have been reviewed and have been found to conform to Commission regulations, regulatory guides, and staff technical positions. These systems have been found to conform to General Design Criteria 2, 4, 5, 19, 34, 44, 45, 46, 55, 56 and 57, to Regulatory Guides 1.26, 1.29, 1.46 and 1.68, and to Branch Technical Position RSB 5-1. The system was found capable of performing its shutdown cooling functions from the control room with only onsite electrical power or offsite power available, assuming the most restrictive single active component failure. It was also found that two independent and redundant barriers are always in place between the RCS and RHR system whenever the RCS pressure is higher than the RHR system design pressure.

"The staff concludes that the design of the systems employed to remove residual heat from the reactor conform to the Commission's regulations and to applicable regulatory guides and staff technical positions, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Desigr. Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

5. 10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

6. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

7. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

8. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

9. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

10. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

11. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," attached to SRP Section 5.4.7.

12. Regulatory Guide 1.29, "Seismic Design Classification."

13. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

14. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

15. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

16. 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

17. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

BRANCH TECHNICAL POSITION RSB 5-1
DESIGN REQUIREMENTS OF THE RESIDUAL HEAT REMOVAL SYSTEM

BACKGROUND

GDC 19 states that, "A control room shall be provided from which actions can be taken to operate the nuclear power unit under normal conditions...."

Normal operating conditions include the shutting down of a reactor; therefore, since the residual heal removal (RHR) system is one of several systems involved in the normal shutdown of all reactors, this system must be operable from the control room.

GDC 34 states that, "Suitable redundance...shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure."

In most current plant designs the RHR system has a lower design pressure than the reactor coolant system (RCS), is located outside of containment and is part of the emergency core cooling system (ECCS). However, it is possible for the RHR system to have different design characteristics. For example, the RHR system might have the same design pressure as the RCS, or be located inside of containment. Plants which may have RHR systems that deviate from current designs will be reviewed on a case-by-case basis. The functional, isolation, pressure relief, pump protection, and test requirements for the RHR system are included in this position.

BRANCH POSITION

A. Functional Requirements

The system(s) which can be used to take the reactor from normal operating conditions to cold shutdown* shall satisfy the functional requirements listed below.

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown* using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

*Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown condition, as described in the Standard Technical Specifications, refers to a subcritical reactor with a reactor coolant temperature no greater than 200°F for a PWR and 212°F for a BWR.

2. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system function can be accomplished, assuming a single failure.

3. The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

4. The system(s) shall be capable of bringing the reactor to a cold shutdown condition , with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure.

B. RHR System Isolation Requirements

The RHR system shall satisfy the isolation requirements listed below.

1. The following shall be provided in the suction side of the RHR system to isolate it from the RCS.

   (a) Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.

   (b) The valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position.

   (c) The valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system.

2. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:

   (a) The valves, position indicators, and interlocks described in item 1(a) - (c),

   (b) One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function, the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

(c) Three check valves in series, or

(d) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

C. Pressure Relief Requirements

The RHR system shall satisfy the pressure relief requirements listed below.

1. To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of the design bases.

2. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not:

   a. Result in flooding of any safety-related equipment.

   b. Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA.

   c. Result in a non-isolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment.

3. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.

D. Pump Protection Requirements

The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system pumps due to overheating, cavitation or loss of adequate pump suction fluid.

E. Test Requirements

The isolation valve operability and interlock circuits must be designed so as to permit on line testing when operating in the RHR mode. Testability shall meet the requirements of IEEE Standard 338 and Regulatory Guide 1.22.

The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The programs for PWRs shall include tests with supporting analysis to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (b) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests.

F. Operational Procedures

The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions.

G. Auxiliary Feedwater Supply

The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure.

H. Implementation

For the purposes of implementing the requirements for plant heat removal capability for compliance with this position, plants are divided into the following three classes:

Class 1 - Full compliance with this position for all plants (custom or standard) for which CP or PDA applications are docketed on or after January 1, 1978. See Table 1 for possible solutions for full compliance.

Class 2 - Partial implementation of this position for all plants (custom or standard) for which CP or PDA applications are docketed before January 1, 1978 and for which an OL issuance is expected on or after January 1, 1979. See Table 1 for recommended implementation for Class 2 plants.

Class 3 - The extent to which the implementation guidance in Table 1 will be backfitted for all operating reactors and all other plants (custom or standard) for which issuance of the OL is expected before January 1, 1979, will be based on the combined I&E and DOR review of related plant features for operating reactors.

TABLE 1. Possible Solutions for Full Compliance with BTP RSB 5-1 and Recommended Implementation for Class 2 Plants

Design Requirements of BTP RSB 5-1	Process and/System or Component/	Possible Solution for Full Compliance	Recommended Implementation for Class 2 Plants (see Note 1)
I. Functional Requirement for Taking to Cold Shutdown a. Capability Using Only Safety Grade Systems b. Capability with either only onsite or only offsite power and with single failure (limited action outside CR to meet SF) c. Reasonable time for cooldown assuming most limiting SF and only offsite or only onsite power	Long-term cooling /RHR drop line/	Provide double drop line (or valves in parallel) to prevent single valve failure from stopping RHR cooling function. (Note: This requirement in conjunction with meeting effects of single failure for long-term cooling and isolation requirements involve increased number of independent power supplies and possibly more than four valves.)	Compliance will not be required if it can be shown that correction for single failure by manual actions inside or outside of containment or return to hot standby until manual actions (or repairs) are found to be acceptable for the individual plant.
	Heat removal and RCS circulation during cooldown to cold shutdown. (Note: Need SG cooling to maintain RCS circulation even after RHRS in operation when under natural circulation) /steam dump valves./	Provide safety-grade dump valves, operators, and power supply, etc. so that manual action should not be required after SSE except to meet single failure.	Compliance required.
	Depressurization /Pressurizer auxiliary spray or power-operated relief valves/	Provide upgrading and additional valves to ensure operation of auxiliary pressurizer spray using only safety-grade subsystem meeting single failure. Possible alternative may involve using pressurizer power-operated relief valves which have been upgraded. Meet SSE and single failure without manual operation inside containment.	Compliance will not be required if a) dependence on manual actions inside containment after SSE or single failure or b) remaining at hot standby until manual actions or repairs are complete are found to be acceptable for the individual plant.
	Boration for cold shutdown /CVCS and boron sampling/	Provide procedure and upgrading where necessary such that boration to cold shutdown concentration meets the requirements of I. Solution could range from (1) upgrading and adding valves to have both letdown and charging paths safety grade and meet single failure to (2) use of backup procedures involving less cost. For example, boration without letdown may be acceptable and eliminate need for upgrading letdown path. Use of ECCS for injection of borated water may also be acceptable. Need surveillance of boron concentration (boronometer and/or sampling. Limited operator action inside or outside of containment if justified.	Same as above.

Design Requirements of BTP RSB 5-1	Process and/System or Component/	Possible Solution for Full Compliance	Recommended Implementation for Class 2 Plants (see Note 1)
II. RHR Isolation	RHR System	Comply with one of allowable arrangements given.	Compliance Required. (Plants normally meet the requirement under existing SRP 5.4.7.)
III. RHR Pressure Relief			
b. Collect and contain relief discharge	RHR System	Determine piping, etc., needed to meet requirement and provide in design.	Compliance will not be required. If it is shown that adequate alternate methods of disposing of discharge are available.
V. Test Requirement			
b. Meet R.G. 1.68. For PWRs, test plus analysis for cooldown under natural circulation to confirm adequate mixing and cooldown within limits specified in EOP.		Run tests and confirming analysis to meet requirement.	Compliance Required.
VI. Operational Procedure			
a. Meet R.G. 1.33. For PWRs, include specific procedures and information for cooldown under natural circulation.		Develop procedures and information from tests and analysis.	Compliance Required.
VII. Auxiliary Feedwater Supply	Emergency Feedwater Supply		
a. Seismic Category I supply for auxiliary FW for at least four hours at hot shutdown plus cooldown to RHR cut-in based on longest time for only onsite or only offsite power and assumed single failure.		From tests and analysis obtain conservative estimate of auxiliary FW supply to meet requirement and provide Seismic Category I supply.	Compliance will not be required. If it is shown that an adequate alternate seismic Category I source is available.

Note 1: The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-grade steam dump valves.

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SECTION 5.4.8 REACTOR WATER CLEANUP SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - None

I. AREAS OF REVIEW

At the construction permit (CP) stage of review, ETSB reviews the information in the applicant's safety analysis report (SAR) in the specific areas that follow. At the operating license (OL) stage of review, the ETSB review consists of confirming the design accepted at the CP stage and evaluating the adequacy of the applicant's technical specifications in these areas.

1. The design of components, design features which influence system availability and reliability, and interconnections with the reactor primary coolant and radwaste systems are reviewed. Fission product removal by the reactor water cleanup system (RWCS) is considered under SRP Section 11.2. The provisions for isolating the RWCS from the reactor system following liquid poison injection, holding filter and demineralizer beds in place if system flow is decreased, straining resins from return flows to the primary system, component venting, and resin transfer are reviewed.

2. The component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability to assure the system capacity will meet the reactor coolant specifications are reviewed.

3. The quality group and seismic design criteria are reviewed.

4. The instrumentation and process controls provided to ensure proper system operation and system isolation when necessary, including instrumentation for (a) automatic system isolation to prevent removal of liquid poison in the event of standby liquid control system actuation and to prevent damage to the filter/demineralizer resins, and (b) monitoring impurity removal (conductivity measurements), differential pressure across pressure-sensitive components, and temperature control prior to demineralization, are reviewed. In addition, the process controls responding to these measurements to maintain operation within the established system parameters are reviewed.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

Secondary reviews are provided as follows: (a) ASB reviews the system design for pipe breaks that could incapacitate safety-related equipment. ASB also reviews the physical separation which is provided between system components and piping to protect essential portions of the system from missiles, pipe whip, and jet impingement that may result from piping breaks. This system is normally inside containment, but the criteria for line breaks outside containment are applied to it (SRP Section 3.5.1); (b) MTEB reviews the material properties and the inservice inspection requirements of the portions of the system that comprise the reactor coolant pressure boundary (SRP Section 5.2.3); (c) ICSB and PSB review instrumentation and components, and power sources with respect to their capacities, capabilities, reliability, and conformance to acceptance standards (SRP Section 7.6); (d) RSB reviews the system drawings of portions of the RWCS that are part of the reactor coolant pressure boundary for correct identification and for capability for isolation from the remainder of the system. RSB evaluates isolation valve performance and verifies that two automatically operated isolation valves in series, or one automatically operated isolation valve and one check valve in series, or one automatically operated isolation valve and one check valve in series, physically separate essential from nonessential portions of the system. RSB also verifies that sufficient instrumentation and controls have been provided to permit plant operators to diagnose and correct system failures that could impair the condition of engineered safety features.

II. ACCEPTANCE CRITERIA

The ETSB will accept the reactor water cleanup system design if the following criteria are met:

1. The reactor water cleanup system should include the following:
   a. Provisions for automatically isolating the RWCS from the reactor coolant system in the event the liquid poison system is actuated for reactor shutdown.
   b. Provisions for automatically isolating the RWCS in the event the nonregenerative heat exchanger effluent temperature exceeds the prescribed resin operating temperature for the cleanup demineralizer resins.
   c. Means for automatically maintaining flow through filter/demineralizer beds in the event of low process flow or loss of process flow through the system, to prevent bed loss. The recirculation loop and holding pump subsystem provided for precoating can serve this purpose if it is activated on loss of flow or low flow conditions.
   d. Means of transferring resins. Sight glass provisions (bull's eyes) are acceptable for monitoring resin transfers. Systems should be designed to prevent "resin traps" in sluice lines. A statement indicating that consideration will be given in the design to avoid resin traps, e.g., a statement that resin transfer lines will be designed to avoid resins collecting in valves, low points, or stagnant areas, will be acceptable for transfer line designs.
   e. Provisions for venting RWCS components through a closed system, i.e., not to the immediate atmosphere. The SAR should state that vent lines run to a ventilation duct exhausting from the plant.

f. Provision, in return lines to the reactor system or condensate system; of resin strainers capable of removing resin particles contained in demineralizer effluents.

2. The system should be capable of maintaining acceptable reactor water purity in normal operation and during anticipated operational occurrences, e.g., reactor startup, refueling, and condensate demineralizer breakthrough. The following points should be included in the system design:
   a. The system should be designed to maintain reactor water purity within the guidelines of Regulatory Guide 1.56. The system should provide demineralization of reactor water through mixed bed resins (beads or powdered) at approximately 1% of the main steam flow rate.
   b. The nonregenerative heat exchangers should be designed to reduce the cleanup flow temperature to the demineralizer operating temperature without the aid of the regenerative heat exchangers.
   c. The RWCS should have the capability to permit processing of excess reactor water during startups, shutdowns and hot standby conditions. Interconnections between the reactor water cleanup and liquid waste and condenstate storage systems to share the processing burden are acceptable.
   d. The RWCS should be designed to permit processing of reactor water during periods of single component failures or equipment downtime.

3. To meet the guidelines of Regulatory Guides 1.26 and 1.29, the portion of the RWCS extending from the reactor vessel and recirculation loops to the outermost drywell isolation valves should be designed to seismic Category I and Quality Group A. The remainder of the system should be designed to Quality Group C and need not be seismic Category I.

4. The RWCS should include provisions for monitoring:
   a. System effluent conductivity. Instrumentation should be consistent with Regulatory Guide 1.56.
   b. Temperature upstream of the demineralizer, to assure the ion exchange resin temperature limits are not exceeded.
   c. Differential pressure, to assure the design limits on filter/demineralizer septums and resin strainers are not exceeded.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this SRP, as may be appropriate for a particular case.

1. ETSB reviews the system description and piping and instrumentation diagrams (P&IDs) to determine the processing sequence, interconnections with other systems, and similarity to systems previously evaluated, and establishes that the following are considered in the applicant's design:

a. Provisions to automatically terminate flow to the RWCS following liquid poison injection into the reactor water.
b. Provisions to automatically terminate flow to the cleanup demineralizers if the nonregenerative heat exchanger effluent temperature exceeds the resin operating temperature limits.
c. Provisions for automatically maintaining flow through filter/demineralizer units in the event system flow decreases to a point where the bed may drop from the septum.
d. Provisions for monitoring resin transfers to assure transfers are complete and design considerations are incorporated to eliminate resin traps.
e. Provisions for venting cleanup system components during drain, fill, and air mixing operations.
f. Provisions for removing resin particles from cleanup system product water to prevent resins from entering the reactor system.

2. ETSB reviews the system capacity and processing flexibility and considers the following:
   a. The process equipment, resin types, and bed volumes compared to those for similar reactors and the RWCS capability compared to the guidelines of Regulatory Guide 1.56.
   b. The design flows and temperatures through the system to assure the criteria for outlet temperature relative to resin temperature are met.
   c. The RWCS capability to process surplus refueling water prior to storage in the refueling water storage tanks or the condensate storage tanks. The system flow rate, surplus capacity in the liquid radwaste system if interconnections exist, and the volume of water to be processed to assure the wastes could be processed in a time which is consistent with the plant requirements, are considered.
   d. Redundant or parallel components which will permit cleanup, if required, during periods of equipment downtime or single component failures.

3. In the review of the quality group and seismic design classification of the system, ETSB compares the design to the guidelines of Regulatory Guides 1.26 and 1.29. In particular, the portion of the RWCS extending from the reactor vessel and recirculation loops to the outermost drywell isolation valves is reviewed to assure conformance to seismic Category I and Quality Group A.

4. ETSB reviews the instrumentation and controls for the reactor water cleanup system to assure that monitors are provided for:
   a. Conductivity of demineralizer effluent.
   b. Temperature and conductivity of demineralizer influent.
   c. Differential pressure across the demineralizer and across the resin strainers.

   ETSB assures that system controls are responsive to the monitor indications to maintain the required temperature and flow and that conductivity meters cover

the entire range up to mandatory shutdown as delineated in the plant technical specifications in the final safety analysis report (FSAR).

IV. EVALUATION FINDINGS

ETSB verifies that sufficient information has been provided and that the review is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The reactor water cleanup system (RWCS) will be used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The scope of the review of the RWCS includes the system capability to meet the anticipated needs of the plant, the capability of the instrumentation and process controls to ensure operation within limits defined in Regulatory Guide 1.56, and the quality group and seismic design classifications contained in Regulatory Guides 1.26 and 1.29. Our review has included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the system design and operation.

"The basis for acceptance in our review has been conformance of the applicant's designs and design criteria to the Commission's regulations and to applicable regulatory guides, as referenced above.

"Based on the foregoing evaluation, we conclude that the proposed reactor water cleanup system is acceptable."

V. REFERENCES

1. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

2. Regulatory Guide 1.29, "Seismic Design Classification."

3. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

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SECTION 5.4.11 PRESSURIZER RELIEF TANK

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Reactor Systems Branch (RSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The pressurizer relief tank is a pressure vessel provided in typical pressurized water reactor (PWR) primary systems to condense and cool the discharge from the pressurizer safety and relief valves. Discharges from small relief valves located inside the containment may also be piped to the tank. Tank capacity is based on a requirement to absorb the pressurizer discharge during a specified step load decrease.

The review of the pressurizer relief tank, as described in the applicant's Safety Analysis Report (SAR), includes the tank, the piping connections from the tank to the pressurizer relief and safety valves, the tank spray system and associated piping, the nitrogen supply piping, and piping leaving the tank to the cover gas analyzer and to the reactor coolant drain tank. The pressurizer relief tank system is nonsafety-related; the review is primarily directed toward assuring that its operation is consistent with transient analyses of related systems and that failure or malfunction of the system could not adversely affect essential systems or components in accordance with applicable criteria.

The review covers the following specific areas:

1. The seismic design classification of the pressurizer relief tank system.

2. The quality standards to which the system will be designed, fabricated, erected, and tested.

3. The measures taken in the design to prevent system performance degradation below acceptable levels as a result of failures of other nearby systems or as a result of the tank failure during an anticipated abnormal occurrence.

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4. The steam condensing capacity of the tank compared to the largest anticipated plant step load decrease.

5. The instrumentation provided to measure and indicate pressurizer relief tank pressure, temperature, and liquid level, and to signal the operator in the event of high or low parameter levels.

6. The tank rupture disk relief capacity compared to the capacity of the pressurizer relief and safety valves.

7. The proposed technical specifications, for operating license applications, as they relate to areas covered in this SRP section.

The review of the pressurizer relief tank system will involve secondary reviews performed by other branches. The results of these reviews are used by ASB to complete overall evaluation of the system. The evaluations performed by others are as follows: the RSB will determine that the anticipated and maximum pressurizer relief and safety valve discharge rates are acceptable based on a review of the limiting transient and will determine that the piping between the valves and the tank is adequately sized. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The MEB will review the operability of components and confirm that the system is designed in accordance with applicable codes and standards.

II. ACCEPTANCE CRITERIA

The design of the pressurizer relief tank system is acceptable if the integrated system design is in accordance with the following criteria:

1. The rupture disks have a relief capacity at least equal to the combined capacity of the pressurizer relief and safety valves with sufficient allowance for rupture disk tolerance.

2. The pressurizer relief tank volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or anticipated abnormal occurrences. The initial temperature of water inside tank should be assumed to be no lower than 120°F.

3. The pressurizer relief tank and rupture disk should be designed for full vacuum so that the collapse of the tank will not occur if the contents are cooled following a discharge of steam without the addition of nitrogen.

4. High temperature, high pressure, high and low liquid level alarms for the pressurizer relief tank have been provided.

5. Regulatory Guide 1.26, as related to the quality group classifications for components and systems.

6. Regulatory Guide 1.29, Item c2, with regard to the location of the tank in relation to other plant systems should be such that the plant safety-related systems would not be endangered in the event of a tank failure.

For those areas of review identified in subsection I as being the responsibility of other branches, the acceptance criteria and their application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures below are used in the construction permit (CP) review to determine that the design criteria and bases and the preliminary design described in the SAR meet the acceptance criteria given in subsection II. For operating license (OL) reviews, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The reviewer selects and emphasizes material from this SRP section, as may be appropriate for a particular case. A determination will be made as to whether the pressurizer relief tank system or portions thereof are safety-related. In confirming this design aspect, an analysis is made in which it is assumed that any system pipe fails or component malfunctions or fails in such a manner as to cause maximum damage to other equipment located nearby. The system will be considered nonsafety-related if its failure does not affect the ability of the reactor facility to achieve and maintain safe shutdown conditions.

1. The SAR is reviewed to establish that the pressurizer relief tank system description and related diagrams clearly delineate system operation and the system capability to accept the steam flow released from the pressurizer for step load decreases. The reviewer examines the adequacy of the design in terms of the seismic design classification (Regulatory Guide 1.29), quality group classification (Regulatory Guide 1.26), and conformance with industry standards. Where necessary, the review will include the requirements for system testing, minimum performance, and surveillance requirements.

2. The SAR is reviewed to determine that the rupture disks on the relief tank have a relief capacity at least equal to the combined capacity of the pressurizer relief and safety valves. The reviewer determines that the tank design pressure provides a conservative margin above the calculated pressure resulting from the maximum design relief and safety valve discharge, i.e., the maximum surge resulting from

complete loss of load. The reviewer verifies that the tank and rupture disks are designed for full vacuum, so as to prevent tank collapse if the contents are cooled following a discharge without nitrogen being added.

3. The pressure suppression capability of the system is reviewed to assure proper system operation. This aspect of the review is similar to the evaluation of the vent clearing and vent flow model for pressure suppression containment systems. The review includes such effects as dynamic loadings and oscillatory behavior of the steam slug in the discharge line. The RSB will verify the mass and energy blowdown data to evaluate the above effects.

4. The piping and instrumentation diagrams are reviewed to verify that high temperature and pressure alarms and high and low liquid level alarms have been provided for the pressurizer relief tank.

5. The reviewer verifies that the system will function following anticipated abnormal occurrences. The reviewer evaluates the failure modes and effects analysis presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR information contains verification that minimum system flow and heat transfer requirements are met for each degraded situation over the required time spans. For each case, the design will be acceptable if minimum system requirements are met.

6. The reviewer determines that failure of the pressurizer relief tank system or portions of the system not designed to seismic Category I, and which are located close to safety-related systems, will not as a result of their failure preclude essential operations of these safety systems. Reference to the general arrangement and layout drawings for structures and systems will be necessary.

7. The reviewer determines that other systems inside containment are protected from the effects of high energy line breaks and moderate energy leakage cracks in the pressurizer relief system. Layout drawings are reviewed to assure that other systems are not located close to the pressurizer relief system, or that protection from the effects of failure will be provided. The means of providing such protection will be described in Section 3.6 of the SAR and the procedures for reviewing this information are given in the corresponding SRP sections.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's Safety Evaluation Report:

"The pressurizer relief tank system includes components and piping such as the pressurizer relief and safety valve connections to the tank, the relief tank spray system piping, the nitrogen supply piping, and piping leaving the tank to the

cover gas analyzer and reactor coolant drain tank. [The review has included the applicant's proposed design criteria and design bases for the pressurizer relief tank system, the adequacy of those criteria and bases, and the requirements for performance of safety-related functions of the system during normal, abnormal, and accident conditions. (CP)] [The review has included the applicant's analysis of the manner in which the design of the pressurizer relief tank and supporting systems conform to the proposed design criteria and design bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the pressurizer relief tank and supporting systems to applicable regulatory guides, branch technical positions, and industry standards.

"The staff concludes that the design of the pressurizer relief tank system conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

2. Regulatory Guide 1.29, "Seismic Design Classification."

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SECTION 6.1.1 ENGINEERED SAFETY FEATURES MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - Containment Systems Branch (CSB)
Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

Engineered Safety Features (ESF) are provided in nuclear plants to mitigate the consequences of design basis or loss-of-coolant accidents, even though the occurrence of these accidents is very unlikely. The General Design Criteria (GDC) 16, 34, 35, 38, 41 and 44 of Appendix A, 10 CFR Part 50, require that certain systems be provided to serve as Engineered Safety Features (ESF). Containment systems, residual heat removal system, emergency core cooling systems, containment heat removal systems, containment atmosphere cleanup systems, and certain cooling water systems are typical of the systems that are required to be provided as ESF.

The steel containment is described in Section 3.8.2 of the Safety Analysis Report (SAR) and reviewed in accordance with Standard Review Plan (SRP) Section 3.8.2. The metallic materials are reviewed in this SRP section.

The emergency core cooling system, the containment heat removal system, the containment cleanup systems and other ESF systems are described in Section 6 of the SAR and are reviewed in accordance with the SRPs for the individual systems. The materials for these systems are reviewed in this SRP.

The cooling system for the reactor auxiliary systems is described in Section 9.2.2 of the SAR and reviewed in accordance with SRP Section 9.2.2. The materials for these systems are reviewed in this SRP.

The areas relating to general materials considerations and ESF fluid chemistry in the design of the engineered safety features are reviewed:

1. Materials and Fabrication

The materials and fabrication procedures used in the engineered safety features are reviewed. The specific areas of review and review procedures are similar to those in Standard Review Plan Section 5.2.3, "Reactor Coolant Pressure Boundary Materials,"

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and to those in Standard Review Plan Section 10.3.6, "Steam and Feedwater System Materials." The purpose of the review is to assure compatibility of the materials with the specific fluids to which the materials are subjected. The review is performed to assure compliance with the General Design Criteria and with the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section II, parts A, B, and C, Section III, Division 1 and 2, and Section IX. Areas that are reviewed include: mechanical properties of materials (including fracture toughness), control of welding procedures, control of low hydrogen welding materials, control of the welding of low alloy steel (including minimum specified preheat), and control of the use of sensitized austenitic stainless steels.

2. Composition and Compatibility of Engineered Safety Features Fluids

The composition of the containment and core spray coolants must be controlled to ensure their compatibility with materials in the containment building, including the reactor vessel, reactor internals, piping, and structural and insulating materials. The methods and procedures to control the chemical composition of solutions recirculated within the containment after design basis accidents (DBA) must be selected (a) to maintain the integrity of the reactor coolant pressure boundary, by preventing stress corrosion cracking of safety-related components, (b) to insure that adequate solution mixing of ESF fluids will occur, and (c) to prevent evolution of excessive amounts of hydrogen within the containment in the unlikely event of a design basis accident.

The time history of the pH of the fluids, including the source and quantity of all soluble acids and bases in the containment after a design basis accident, is reviewed.

Containment and core spray solutions must be stable under long-term storage conditions and during prolonged operation of the sprays. Some of these solutions contain boron for reactivity control and other additives (such as thiosulfates) for reacting with gaseous fission products.

In many instances the ESF coolant solutions are stored in more than one form (such as boric acid solution and a sodium hydroxide solution) and mixed only when the ESF are called upon to operate during an emergency. In some plants, the coolant is stored as a boric acid solution that is neutralized by (dry) sodium phosphates mounted in baskets inside the containment after the ESF sprays are activated.

The controls on contaminants, such as chlorides, lead, zinc, sulfur, or mercury, in the ESF fluids are reviewed. Potential sources of these contaminants, such as coatings and nonmetallic thermal insulation, that will be exposed to ESF fluids in DBA environments are evaluated.

Peeling, flaking or delamination of coatings can result in clogging of ESF system strainers and spray nozzles and thereby stop or slow down the flow rates of the ESF fluids. The qualification program for the protective coatings is reviewed.

The Containment Systems Branch (CSB) is responsible for reviewing hydrogen buildup in the containment as described in SRP Section 6.2.5, "Combustible Gas Control in Containment." The Materials Engineering Branch (MTEB) reviews only corrosion and hydrogen evolution rates as submitted in the SAR.

The procedures, methods, and liquid pathways used in the control of pH, conductivity, and chlorides are reviewed by the Accident Analysis Branch (AAB).

MTEB has secondary review responsibility for corrosion and integrity of Containment Materials as described in SRP Section 3.8.2.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Materials and Fabrication

Materials for use in ESF must be selected for their compatibility with ECCS and containment spray solutions and other ESF fluids. The materials specified for use in these systems must be as given in Appendix I to Section III of the Code, and parts A, B and C of Section II of the Code. Fracture toughness of the materials shall be as stated in SRP Section 10.3.6, Subsection II.1. Cold-worked austenitic stainless steels must have a maximum 0.2% offset yield strength of 90,000 psi to reduce the probability of stress corrosion cracking in these systems.

Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials," describes acceptable Code cases that may be used in conjunction with the above specifications.

Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," describes acceptable criteria for preventing intergranular corrosion of stainless steel components of the ESF. Furnace-sensitized material should not be allowed in the ESF, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling and storage, for testing the materials prior to fabrication, and for ensuring that no deleterious sensitization occurs during welding.

Components and systems are to be cleaned in conformance with the requirements of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

Branch Technical Position MTEB 5-7, "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," describes acceptable criteria for the use of austenitic stainless steel piping in boiling water reactors. (See SRP Section 5.2.3.)

Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," describes acceptable criteria for assuring the integrity of welds in austenitic stainless steel ESF components. The control of delta ferrite content of weld filler metal is specified in this guide, which sets forth an acceptable basis for delta ferrite content of weld filler metal.

The acceptance criteria for ferritic steel welding are as follows:

a. The amount of minimum specified preheat must be in accordance with the recommendations of the Code, Section III, Appendix D, Article D-1000, and Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel," unless an alternate procedure is justified.

b. Moisture control on low hydrogen welding materials shall conform to the requirements of the Code, Section III, Articles NB, NC, ND-2000 and 4000, and AWS D1.1, "Structural Welding Code," unless alternate procedures are justified.

c. For areas of limited accessibility, the criteria of SRP Section 10.3.6.II.2.c shall apply.

The composition of nonmetallic thermal insulation for components of ESF should be controlled as described in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Concentrations of leachable contaminants and added inhibitors should be controlled as specified in position C.2.b and Figure 1 of this guide to reduce the probability of stress corrosion cracking of austenitic stainless steel components.

The use of nonmetallic insulation on nonaustenitic stainless steel components should be controlled as above. The moisture dripping from wet insulation on any component can affect austenitic stainless steel that is at a physically lower elevation.

The criteria for coatings to be used in containments are described in Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Identified quantities of soluble acids and bases within the containment must not be great enough to cause excessive hydrogen generation or deleterious corrosion.

2. Composition and Compatibility of Engineered Safety Features Fluids

The qualification program for coating systems should confirm that the systems used on ESF will not possibly stop or slow down the flow rates of the ESF fluids during a design basis accident. The compositions of containment spray and core cooling water for PWRs should be controlled to ensure a minimum pH of 7.0, as given in Branch Technical Position MTEB 6-1 attached to this SRP section. Experience has shown that maintaining the pH of borated solutions at this level will help to

inhibit initiation of stress corrosion cracking of austenitic stainless steel components.

Hydrogen generation results from the corrosion of materials by the containment sprays during design basis accident. Hydrogen release within the containment should be controlled as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

a. Pressurized Water Reactors (PWRs)

The hydrogen generation from the corrosion of materials within containment, such as aluminum and zinc, depends upon the corrosion rate which in turn depends upon such factors as the coolant chemistry, the coolant pH, the metal and coolant temperature, and the surface area exposed to attack by the coolant.

The reviewer compares the assumed corrosion rates of materials in containment with standard corrosion rate data.

b. Boiling Water Reactors (BWRs)

Water used in the engineered safety feature systems should be controlled to provide assurance against stress corrosion cracking of unstabilized austenitic stainless steel components. Water used for emergency core cooling systems and spray systems should be controlled to ensure the following limits:

Conductivity = 3 to 10 μmhos/cm @ 25°C
Chloride (Cl^-) < 0.50 ppm
pH = 5.3 to 8.6 @ 25°C

Hydrogen generation in BWR containments is assumed to follow the same characteristics as in PWRs in that the rates of hydrogen generation will rise with increasing zinc corrosion as the temperature rises, and will change with any change in pH.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

To ascertain that the acceptance criteria given in Section II are met, the reviewer examines each of the review areas given in Section I for the required information, using the following procedure:

1. Materials and Fabrication

The reviewer examines the information on the compatibility of the ESF materials of construction with the ESF fluids to verify that all materials used are compatible.

The reviewer considers the composition of the sprays and any mixing processes that might occur during operation of the sprays.

The reviewer verifies that the materials proposed for the ESF are in conformance with Appendix I of Section III of the Code, and with parts A, B and C of Section II of the Code. He verifies that cold-worked austenitic stainless steels used in fabrication of the ESF are in conformance with Section II.1.

The reviewer verifies that the controls of ferritic steel welding are in conformance with Subsection II.1. The reviewer verifies that the fracture toughness of the materials is in accordance with the requirements of the Code.

The methods of controlling sensitized stainless steel in the ESF systems are examined by the reviewer who verifies that the methods are in conformance with Regulatory Guide 1.44. This applies especially to the cleaning and protection of stainless steel from contamination during handling and storage, to the verification of nonsensitization of the materials, and to the qualification of welding procedures using ASTM A262. If alternative methods of testing the qualification welds for degree of sensitization are proposed by the applicant, the reviewer determines if these are satisfactory, based on the degree to which the alternate methods provide the needed results.

The methods for controlling the amount of delta ferrite in stainless steel weld deposits are examined by the reviewer in accordance with Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The reviewer determines whether nonmetallic thermal insulation will be used on components of the ESF, and, if it is, he verifies that the amount of leachable impurities in the specified insulation lie within the "acceptable analysis" area of Figure 1 of Regulatory Guide 1.36, as discussed in the acceptance criteria, Section II.1.

The reviewer verifies that the coatings used in the containment conform with Regulatory Guide 1.54.

2. Composition and Compatibility of Engineered Safety Features Fluids

a. Pressurized Water Reactors (PWRs)

The reviewer determines that the coolant spray will have a minimum pH of 7.0 and reviews the methods of ascertaining that the pH will remain above this minimum during the operation of the sprays. The reviewer examines the control of pH of such coolants to evaluate the short-term (during the mixing process) compatibility and long-term compatibility of these sprays with all safety-related components within the containment.

The AAB reviews the chemical and radiation environment discussed in SRP Section 3.11 Appendix, "Chemical and Radiological Environment in Containment During Postulated Accidents" and in SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System." The AAB examines the paths that the solutions would follow in the containment from sprays and emergency core cooling systems to the sump, for both injection and recirculation phases to verify that no areas accumulate very high or low pH solutions and that any assumptions regarding pH in the modeling of containment spray fission product removal are valid.

The reviewer examines the methods of storing the ESF fluids to determine whether deterioration will occur either by chemical instability or by corrosive attack on the storage vessel. The reviewer determines what effects such deterioration could have on the compatibility of these ESF coolants with both the ESF materials of construction and the other materials within the containment.

b. Boiling Water Reactors (BWRs)

The reviewer verifies that the chemistry of the water used for the emergency core cooling systems and the containment spray systems is controlled to the limits given in Subsection II.2.b.

IV. EVALUATION FINDINGS

The reviewer verifies that the information provided supports conclusions of the following type, which are to be included in the staff's safety evaluation report:

"The materials selected for the Engineered Safety Features satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi. Fracture toughness of the ferritic materials meets the requirements of the Code.

"The controls on the pH and chemistry of the reactor containment sprays and the Emergency Core Cooling water following a postulated loss-of-coolant or design basis accident, are adequate to reduce the probability of stress corrosion cracking of the austenitic stainless steel components and welds of the Engineered Safety Features systems in containment throughout the duration of the postulated accident to completion of cleanup. The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the requirements of Regulatory Guide 1.31, 'Control of Ferrite Content of Stainless Steel Weld Metal', and Regulatory Guide 1.44, 'Control of the Use of Sensitized Stainless Steel'. Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the Engineered Safety Features are in accordance with Regulatory Guide 1.36, 'Nonmetallic Thermal Insulation

for Austenitic Stainless Steel". The control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, is in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident", and provides assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment. The protective coating systems have been qualified by tests acceptable to the staff. This qualification provides reasonable assurance that the coating systems will not degrade the operation of the ESF by delaminating, flaking or peeling. Conformance with the Codes and Regulatory Guides and with the staff positions mentioned above, constitute an acceptable basis for meeting in part the requirements of General Design Criteria 16, 34, 35, 38, 41 and 44."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criteria.

2. ASME Boiler and Pressure Vessel Code, Section II, Parts A, B, and C, Section III, Division 1, including Appendix I, Section III, Division 2, and Section IX, American Society of Mechanical Engineers.

3. ASTM A-262, "Detecting Susceptibility to Intergranular Attack in Stainless Steel," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

4. AWS D1.1, "Structural Welding Code," American Welding Society.

5. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

6. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

7. Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

8. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

9. Regulatory Guide 1.44, "Control of the Use of Sensitized Steel."

10. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel."

11. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants."

12. Standard Review Plan 3.11, Appendix, "Chemical and Radiological Environment in Containment During Postulated Accidents."

13. Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials."

14. Standard Review Plan 6.2.5, "Combustible Gas Control in Containment."

15. Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System."

16. Standard Review Plan 10.3.6, "Steam and Feedwater Systems Materials."

17. Branch Technical Position MTEB 5-7, "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

18. Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water," appended.

BRANCH TECHNICAL POSITION MTEB 6-1
pH FOR EMERGENCY COOLANT WATER FOR PWRs

A. Background

To establish the minimum value of pH in post-accident containment sprays in Pressurized Water Reactors, the Materials Engineering Branch has reviewed the available information and recommended the criteria listed in the Branch Technical Position below.

The minimum pH value of 7.0 follows from the Westinghouse report (Ref. 1) conclusion that, in ECCS solutions adjusted with NaOH to pH 7.0 or greater, no cracking should be observed at chloride concentrations up to 1000 ppm during the time of interest. Figure 7 of the Westinghouse report shows that time for initiation of cracking of sensitized and nonsensitized U-bend specimens of Type 304 austenitic stainless steel in solutions of 7.0 pH having 100 ppm chloride was seven and one half months and ten months, respectively. These time periods are more than ample time to allow cleanup after the hypothetical design basis accident.

The great majority of tests reported in the Oak Ridge report, Reference 2, were performed with pH of 4.5, and only two tests were conducted with pH values other than 4.5. Some cracking was observed at pH 7.5 in the sensitized 304 stainless steel U-bend specimens after two months exposure to pH 7.5 and chloride concentration of 200 ppm. All of the 316 stainless steel specimens showed no evidence of cracking. Considering the fact that in U-bend specimens the material was sensitized, stressed beyond yield, and plastically deformed, we conclude that the reported test conditions were much more severe than the stress conditions likely to exist in the post-accident emergency coolant systems.

We agree with the Oak Ridge conclusion that absolute freedom from failure of any complex system such as a spray system can never be guaranteed, but, by proper design, fabrication, and control of the corrosive environment, the probability of failure can be significantly reduced. Our recommended minimum pH of seven is somewhat higher than the Oak Ridge recommendation of a minimum of 6.5.

B. Branch Technical Position

MTEB criteria for pH level of post-accident emergency coolant water to reduce the probability of stress-corrosion cracking of austenitic stainless steel components, nonsensitized or sensitized, non-stressed or stressed, are as follows:

1. Minimum pH should be 7.0.

2. The higher the pH (in the 7.0 to 9.5 range), the greater the assurance that no stress corrosion cracking will occur.

3. If a pH greater than 7.5 is used, consideration should be given to the hydrogen generation problem from corrosion of aluminum in the containment.

C. References

1. D. D. Whyte and L. F. Picone, "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Environment," WCAP-7798-L, Westinghouse Nuclear Energy Systems, November 1971 (NES Proprietary Class 2).

2. J. C. Griess and E. E. Creek, "Design Considerations of Reactor Containment Spray Systems - Part X, The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions," ORNL-TM-2412, Part X, Oak Ridge National Laboratory, May 1971.

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SECTION 6.1.2 ORGANIC MATERIALS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

1. The coating systems (paints) used inside the containment are evaluated as to suitability for design basis accident (DBA) conditions.

2. The stability of materials (particularly organics) and their decomposition products are examined to determine the potential for interactions with engineered safety features (ESF). Radiation and chemical effects are considered.

The radiation and chemical environments under DBA conditions are reviewed under Standard Review Plan Section 3.11. The fission product removal effectiveness of the containment spray system is reviewed under Standard Review Plan Section 6.5.2.

II. ACCEPTANCE CRITERIA

A coating system to be applied inside a containment is acceptable if:

1. It meets Regulatory Guide 1.54 (Ref. 1) and ANSI N101.2 (Ref. 2) or equivalent.

2. No adverse interactions with engineered safety features are likely as a result of materials released by radiation decomposition or chemical reaction of the coating system in the containment post-accident environment.

In particular, under normal and DBA conditions:

a. A combustible gas mixture will not result.

b. Organic iodide formation will not exceed that assumed in Regulatory Guides 1.3 and 1.4 (Refs. 3 and 4).

c. No solid debris that can fall into the containment sump are generated, unless it can be justified that the debris will not adversely affect the performance of post-accident fluid systems.

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III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on the areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether or not it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

At the construction permit review stage, the reviewer verifies that the applicant has committed to using protective coating systems which meet the acceptance criteria.

At the operating license review stage, the reviewer determines the types and quantities of radiation and chemical decomposition products that can be produced from all the paints and organic materials which are exposed to the containment atmosphere. The paints and organic materials to be considered include those paints that are specified in the Safety Analysis Report (SAR), unspecified protective coatings on small machinery and equipment, and organic materials such as uncovered cable insulation. The determination is based on documented test data provided by the applicant. If test data are unavailable, a conservative analysis is required. The environmental conditions for the test and analysis must be comparable to those specified in Section 3.11 of the SAR.

If combustible gases such as hydrogen and methane can be generated, the reviewer notifies the Containment Systems Branch if this source is not included in Section 6.2.5 of the SAR. If a system to control combustible vapors is not provided, then the release of volatile alkanes to form organic iodides is of additional concern. The yield of organic iodides relative to the total iodine released after a DBA is estimated using the data of Reference 5 and any applicable experimental results submitted by the applicant. The formation of organic iodides should not exceed that assumed in Regulatory Guides 1.3 and 1.4.

If solid debris can be produced and can reach the containment recirculation sump, the Containment Systems Branch and Reactor Systems Branch should be notified in order that the effects of the debris on the operation of post-accident fluid systems may be assessed.

Any exception to Regulatory Guide 1.54 involving quality assurance and quality control requirements should be referred to the Quality Assurance Branch for review and resolution.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following types, to be included in the staff's safety evaluation report:

> "The containment coating systems have been evaluated as to their suitability to withstand a postulated design basis accident (DBA) environment. The coating systems chosen by the applicant have been qualified under conditions which take into account the postulated DBA conditions. No adverse interactions (under DBA conditions) between the decomposition products and the engineered safety features have been established. The amount of unqualified paint in the containment is not significant.

The staff concludes, therefore, that the coating system will not adversely affect the consequences of postulated design basis accidents and is acceptable."

V. REFERENCES

1. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants."

2. ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," American National Standards Institute (1972).

3. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

4. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

5. A. K. Postma and R. W. Zavadoski, "Review of Organic Iodide Formation Under Accident Conditions in Water-Cooled Reactors," WASH-1233 (1972).

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SECTION 6.2.1 CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - See secondary review responsibilities of the seven SRP sections listed below for the various containment types and aspects.

INTRODUCTION

The CSB reviews information regarding the functional capability of the reactor containment presented in Section 6.2.1 of the applicant's safety analysis report (SAR). The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

The design and sizing of containment systems are largely based on the pressure and temperature conditions which result from release of the reactor coolant in the event of a loss-of-coolant accident (LOCA). The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the emergency core cooling system cools the reactor core.

The evaluation of a containment functional design includes calculation of the various effects associated with the postulated rupture in the primary or secondary coolant system piping. The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy are determined from a solution of the incremental space and time-dependent energy, mass, and momentum equations. The basic functional design requirements for containment are given in General Design Criteria 16 and 50 in Appendix A to 10 CFR Part 50. General Design Criterion 50, among other things, requires that consideration be given to the potential consequences of degraded engineered safety features, such as the containment heat removal system and the emergency core cooling system, the limitations in defining accident phenomena, and the conservatism of calculational models and input parameters, in assessing containment design margins.

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There are a number of different containment types and designs, and several aspects of containment functional design that are within the scope of SAR Section 6.2.1. The various containment types and aspects to be reviewed under this SRP section have been separated and assigned to a set of other SRP sections as follows:

a. Pressurized water reactor (PWR) dry containments, including subatmospheric containments (SRP section 6.2.1.1.A).
b. Ice condenser containments (SRP section 6.2.1.1.B).
c. Mark I, II, and III boiling water reactor (BWR) pressure-suppression type containments (SRP section 6.2.1.1.C).
d. Subcompartment analysis (SRP section 6.2.1.2).
e. Mass and energy release analysis for postulated loss-of-coolant accidents (SRP section 6.2.1.3).
f. Mass and energy release analysis for postulated secondary system pipe ruptures (SRP section 6.2.1.4).
g. Minimum containment pressure analysis for emergency core cooling system (ECCS) performance capability studies (SRP section 6.2.1.5).

A separate SRP section has been prepared for each of these areas.

Areas related to the evaluation of the containment functional capability are treated in other SRP sections; e.g., containment heat removal (SRP section 6.2.2), combustible gas control (SRP section 6.2.5), and containment leakage testing (SRP section 6.2.6).

I. AREAS OF REVIEW

The items reviewed are described in the "Areas of Review" subsections of the seven SRP sections listed above.

II. ACCEPTANCE CRITERIA

The acceptance criteria are given in the "Acceptance Criteria" subsections of the seven SRP sections listed above.

II. REVIEW PROCEDURES

Review procedures are given in "Review Procedures" subsections of the seven SRP sections listed above.

IV. EVALUATION FINDINGS

The results of the reviews under the seven SRP sections listed above are consolidated into a single set of findings. The reviewer verifies that sufficient information has been provided and that his evaluation is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

Containment Functional Design

The scope of review of the functional design of the containment for the _____ nuclear power plant has included a review of plant arrangement drawings, system drawings, and descriptive information for the containment building, subcompartments, and associated systems, components, and structures that are essential to the functional capability and

integrity of the containment. The review has included the applicant's proposed design bases for the containment building and internal structures, and associated structures and systems upon which the containment function depends, and the applicant's analysis of postulated accidents and operational occurrences which support the adequacy of the design bases.

"The basis for the staff's acceptance has been conformance of designs and design bases for the containment building, internal structures, and associated systems, components, and structures to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of containment structures, systems, and components should be discussed.)

"The staff concludes that the containment functional design conforms to applicable regulations, guides, staff positions, and industry standards, and is acceptable."

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3. ASME Boiler and Pressure Vessel Code, Section II, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

4. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

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15. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

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17. "FLOOD/MOD001 - A Code to Determine the Core Reflood Rate for a PWR Plant with Two Core Vessel Outlet Legs and Two Core Vessel Inlet Legs," Interim Report, Aerojet Nuclear Company, October 11, 1972.

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NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.1.1.A PWR DRY CONTAINMENTS, INCLUDING SUBATMOSPHERIC CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Instrumentation and Control Systems Branch (ICSB)

I. AREAS OF REVIEW

For pressurized water reactor (PWR) plants with dry containments, the CSB review covers the following areas:

1. The temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks.

2. The maximum expected external pressure to which the containment may be subjected.

3. The minimum containment pressure used in analyses of emergency core cooling system capability.

4. The effectiveness of static and active heat removal mechanisms.

5. The pressure conditions within subcompartments and acting on system components and supports due to high energy line breaks.

6. The instrumentation provided to monitor and record containment conditions during and following an accident.

7. The proposed technical specifications at the operating license stage of review pertaining to the surveillance requirements for spring or weight loaded check valves used in subatmospheric containments, and vacuum relief devices.

At the request of the CSB, the ICSB will evaluate; (1) the effectiveness of the administrative controls and the instrumentation and control provisions to prevent inadvertant peration of the containment heat removal systems or system trains; and (2) the functional capability of the post-accident monitoring instrumentation and recording equipment.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

II. ACCEPTANCE CRITERIA

The following acceptance criteria complement General Design Criterion 50 and apply to the design and functional capability of PWR dry containments:

1. For plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break.

2. For plants at the operating license (OL) stage of review, the peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break, should be less than the containment design pressure. In general, the peak calculated containment pressure should be approximately the same as at the construction permit stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin.

3. The containment pressure should be reduced to less than 50% of the peak calculated pressure for the design basis loss-of-coolant accident within 24 hours after the postulated accident. This is in conformance with the requirement of General Design Criteria 38 "Containment Heat Removal Systems," that the containment pressure and temperature following a loss-of-coolant accident be rapidly reduced. If analysis shows that the calculated containment pressure may not be reduced to 50% of the peak calculated pressure within 24 hours the AAB should be notified.

4. For subatmospheric containments, the containment pressure should be reduced to below atmospheric pressure within one hour after the postulated accident, and the subatmospheric condition maintained for at least 30 days.

5. The loss-of-coolant accident analysis should be based on the assumption of loss of offsite power and the most severe single active failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.

6. The containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single active failure in the containment heat removal systems (e.g., a fan, pump, or valve failure) or the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis.

7. The minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").

8. Provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems; or for steel containment vessels, vacuum relief devices provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE (Ref. 3), and applicable requirements of General Design Criteria 54 and 56.

9. If the primary containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.

10. Containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments (See SRP Section 6.2.1.2, "Subcompartment Analysis").

11. Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Regultory Guide 1.97, "Instrumentation For Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following An Accident," should be followed.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches. The acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The following procedures are for the review of PWR dry containments. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis for aspects of functional design common to a class of dry containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I of this plan. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The CSB reviews the containment response analyses to determine the acceptability of the calculated containment design pressure and temperature, and in addition, the containment depressurization time. The AAB must be notified if the containment depressurization time does not meet the acceptance criterion). The CSB reviews the assumptions made in the analyses to maximize the calculated containment pressure and temperature. The CSB determines the conservatism of the respective containment response analyses by comparing the analytical models, and the assumptions made, with the acceptance criteria in subsection II, and by performing appropriate confirmatory analyses. It is not necessary to perform accident pressure calculations for every plant. The CSB will ascertain, however, that the adequacy of the applicant's calculational model has been demonstrated. The CSB determines that the applicant has identified the pipe break(s) resulting in the highest containment pressure and temperature. Hot leg, cold leg (pump suction), and cold leg (pump discharge) pipe breaks of the reactor coolant system, and secondary system steam and feedwater line breaks, should be analyzed by the applicant. The CSB reviews the assumptions used to determine that the analyses are acceptably conservative.

The CSB performs confirmatory containment response analyses when necessary using the CONTEMPT-LT computer code (See References 7, 8, and 9 for a description of this code). The purpose of these analyses is to confirm the applicant's predictions of the response of the containment to loss-of-coolant accidents and main steam and feedwater line breaks. In general, only the limiting pipe breaks, i.e., the pipe breaks which establish the containment design pressure and containment depressurization time, are analyzed. However, if in the judgment of the CSB the worst break has not been identified, other pipe breaks will be analyzed.

The CSB reviews analyses of the external pressure of the containment structure caused by pressure and temperature changes inside the containment due to inadvertent operation of containment heat removal systems. The CSB determines whether the most severe condition has been identified, and whether the analysis was done in a conservative manner. The CSB evaluates the acceptability of the provisions made in the plant design to mitigate or withstand the consequences of the above postulated events, and evaluates in conjunction with the ICSB, the administrative controls and instrumentation and control provisions to preclude these events.

The CSB determines whether instrumentation capable of withstanding the post-accident environment, and recording equipment, and has been provided to monitor and record the course of an accident within the containment. The CSB also determines whether the instrumentation and recording equipment can accomplish the objectives stated in Regulatory Guide 1.97. This review is coordinated with the ICSB. The ICSB, under SRP section 7.3, has review responsibility for the acceptability of, and the qualification test

program for the sensing and actuation instrumentation of the plant protection system and the post-accident monitoring instrumentation and recording equipment.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in SRP Section 6.2.1.

V. REFERENCES

The references for this SRP section are listed in SRP Section 6.2.1.

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 6.2.1.1.B ICE CONDENSER CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Instrumentation and Control Systems Branch (ICSB)
Accident Analysis Branch (AAB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The CSB review of ice condenser containments includes the following areas:

1. The pressure and temperature conditions in the containment due to a spectrum (including break size and location) of loss-of-coolant accidents (i.e., reactor coolant system pipe breaks) and steam and feedwater line breaks.

2. The maximum expected external pressure to which the containment may be subjected.

3. The design and qualification testing of ice condenser components.

4. The pressure conditions within containment internal structures and acting on system components and supports due to high energy line breaks.

5. The maximum allowable operating deck steam bypass area for a full spectrum of reactor coolant system pipe breaks.

6. The design provisions and proposed surveillance program to assure that the ice condenser will remain operable for all plant operating conditions.

7. The design and qualification testing of the return air fan systems and system components.

8. The effectiveness of static and active heat removal mechanisms.

9. The minimum containment pressure used in the analyses of emergency core cooling system capability.

10. The instrumentation provided to monitor and record containment atmosphere pressure and temperature and sump water temperature under post-accident conditions.

USNRC STANDARD REVIEW PLAN

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

11. The proposed technical specifications, at the operating license stage of review, pertaining to the surveillance requirements for steam bypass area, return air fan system operability, ice condenser operability, and vacuum relief devices.

The structural and mechanical design adequacy of the containment, its internal structures and mechanical components are the review responsibility of the SEB and the MEB. The adequacy of the containment response to assumed piping system ruptures is the responsibility of the CSB. The ICSB has review responsibility for the ice condenser containment design. The fission product removal capability of the ice condenser is evaluated by AAB under SRP Section 6.5.4.

II. ACCEPTANCE CRITERIA

The following acceptance criteria apply to the design and functional capability of ice condenser containments:

1. The ice condenser components should be designed, fabricated, erected, and tested in accordance with Group B quality standards, as recommended by Regulatory Guide 1.26.

   The ice condenser components should be designated Category I (seismic); i.e., designed to withstand the effects of the safe shutdown earthquake without loss of function, as recommended by Regulatory Guide 1.29.

   Analyses or requalification tests should be performed for all ice condenser components that are changed in design from that reported in Appendices M and N to the D.C. Cook FSAR (Ref. 27) to assure that the ice condenser will remain operable in the accident environment for as long as accident conditions require. If a component was originally qualified by analytical methods, confirmation of the new design by reanalyses or a test program will be acceptable. For components that were originally qualified by a test program, the redesigned component should be requalified by a test program.

2. The containment pressure and temperature response to postulated loss of coolant accidents should be calculated using the LOTIC-1 (or an equivalent) computer code (Ref. 34).

   For plants under review for construction permits, the containment design pressure should provide at least a 20% margin above the highest calculated accident pressure. For plants under review for operating licenses, the highest calculated accident pressure should not exceed the design pressure of the containment.

3. The containment pressure and temperature response to postulated secondary system pipe ruptures should be based on the most severe single active failure of the isolation provisions in the secondary system (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions

selected should result in the highest calculated containment pressure or temperature, depending on the purpose of the analysis.

4. Ice condenser subcompartment or control volume differential (internal) pressures should be calculated using the Transient Mass Distribution (TMD) computer code (Ref. 28), without the augmented critical flow correlation. TMD should incorporate the heat transfer correlation developed from the 1974 full scale ice condenser tests and should include the compressibility factor "Y" in the incompressible flow equation.

   For plants being reviewed for construction permits, the design differential pressures for all ice condenser control volumes or subcompartments, and system components (e.g., reactor vessel, pressurizer, steam generators) and supports, should provide at least a 40% margin above the highest calculated differential pressures. For plants being reviewed for operating licenses, the highest calculated differential pressures for all ice condenser control volumes or subcompartments should not exceed the corresponding design differential pressures.

   The operating deck, steam generator and pressurizer enclosures, and ice condenser lower inlet doors should be designed to withstand the maximum calculated reverse differential pressures between the upper and lower compartments using the LOTIC-2 computer code (Ref. 35). To account for uncertainties in the analysis of reverse differential pressures, an adequate margin should be provided above the maximum calculated reverse differential pressure.

5. The maximum allowable area for steam bypass of the ice condenser should be greater than the identifiable bypass area for the plant (e.g., the drainage provisions to allow containment spray water to return from the upper compartment to the sumps in the lower compartment). The bypass area capability of the plant should be based on analyses of the spectrum of postulated reactor coolant system pipe breaks, and should be about 35 square feet or greater.

6. The design of the ice condenser system should incorporate provisions for periodic inservice inspection and testing of essential system components; e.g., the ice baskets and doors, the ice condenser temperature monitoring system, and the available mass of ice. The inspection and test program should assure the integrity and operability of the ice condenser system and should satisfy the requirements of General Design Criteria 39 and 40.

7. The return air fan system components should be designed, fabricated, erected, and tested in accordance with Group B quality standards, as recommended by Regulatory Guide 1.26. The system should be designated Category I (seismic) as recommended by Regulatory Guide 1.29.

The inservice inspection and testing program for the return air fan system should satisfy the requirements of General Design Criteria 39 and 40.

Analyses or tests should be performed for the return air fan system components to demonstrate that the system will remain operable in the accident environment for as long as accident conditions require.

8. Inadvertent operation of engineered safety features (e.g., the return air fan system or the containment spray system) should not cause the external design pressure of the primary containment to be exceeded. This may be accomplished through conservative containment design, use of vacuum relief devices, or electrical interlocks that preclude inadvertent operation of the spray and fan systems. Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE (Ref. 3) and should meet applicable requirements of General Design Criteria 54 and 56.

9. Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature, and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," should be followed.

10. The minimum calculated containment pressure as determined by the LOTIC-2 Code (Ref. 35) should not be less than that used in the analysis of the emergency core cooling system capability (See SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").

11. For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of ice condenser containments. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis for aspects of functional design common to a class of ice condenser containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I of this plan. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

1. The CSB evaluates the design of the ice condenser type containment by comparing it to the design information presented in Appendices M and N to the D.C. Cook FSAR, and discussed in the staff's safety evaluation report on the plant (Ref. 27). The CSB has reviewed the design of the Cook ice condenser as reported in these documents and has found that it satifies the acceptance criteria stated in subsection II. Any differences from the design reported in the Cook documents are evaluated. The CSB determines that all design changes have been justified, and the components have been requalified for use in the ice condenser by the same methods originally used to qualify them, i.e., for simple structures which were qualified by analytical methods, a reanalysis will be accepted; and for components qualified by test programs, the tests should be operated on the revised design.

   The CSB compares the quality standards applied to the ice condenser to Regulatory Guide 1.26.

   The CSB compares the seismic design classification of the ice condenser to Regulatory Guide 1.29.

2. The CSB reviews the analysis of the containment pressure and temperature response to postulated loss-of-coolant accidents. The CSB has reviewed the LOTIC-1 code which is used to determine the containment pressure and temperature response, and has determined that the code is acceptable for containment analysis. The CSB assures that the LOTIC-1 code has been used and that the input assumptions to the code are conservative. Code revisions and improvements will also be considered.

   The CSB reviews the analysis of the containment temperature and pressure response to postulated secondary system pipe ruptures. The CSB determines the acceptability of the analytical methods and plant sensitive input assumptions used in the analysis of the containment response.

   CSB determines from the results of analyses of postulated loss-of-coolant accidents and secondary system pipe ruptures that the peak calculated containment pressure does not exceed the design pressure of the containment, for plants at the operating license stage of review. For plants at the construction permit stage of review, the CSB will ascertain from the results of anlyses reported in the safety analysis report that the design pressure provides a margin of at least 20% above the maximum calculated pressure.

   The CSB determines that the maximum temperature transients calculated for postulated loss-of-coolant accidents and secondary system pipe ruptures have been considered in establishing the environmental qualification requirements for equipment and components required to mitigate the consequences of loss-of-coolant accidents and secondary system pipe ruptures, respectively. If thermal analysis is used to establish the qualification of instrumentation and components for use in superheated

steam environments, upon request of the PSB or ICSB, the CSB will review the analytical methods and assumptions used in the analysis. The CSB will inform the requesting branch(es) of the acceptability of the thermal analyses.

The CONTEMPT-4 code is being developed to provide improved capability to analyze the long term response of an ice condenser containment to a loss-of-coolant accident. When the CONTEMPT-4 code is available, the CSB will perform confirmatory analyses using this code.

3. The TMD code is used to evaluate the transient pressure responses (internal) of the ice condenser containment subcompartments. The code is described in the proprietary report WCAP-8077 (Ref. 28). The TMD code utilizes an ice condenser heat transfer coefficient obtained from the 1974 full-scale section tests of the ice condenser. The TMD code also utilizes a compressibility factor "Y" to account for compressible flow effects. As stated in the D.C. Cook Safety Evaluation Report, the CSB has reviewed the assumptions and equations used in the TMD code and with the exception of the critical flow model used to predict subcompartment vent mass flow rates, has concluded that the TMD code conservatively calculates transient pressure response.

   The TMD code calculates the critical flow of a two-component, two-phase fluid (air, steam, and water) assuming a thermal equilibrium condition. However, a correction factor is them applied to the calculated critical flow. The CSB has not accepted the use of this corrected critical flow, referred to as "augmented flow," and has required that the short-term transient responses of subcompartments be determined using the TMD code without applying a correction factor to the critical flow; i.e., without the "augmented flow" correlation.

   Before accepting the containment transient responses calculated by the TMD code, the CSB reviews the modeling of the containment subcompartments, the size and area of assumed vents between nodes, volumes of nodes, the flow loss coefficients for each vent modeled, and the heat transfer coefficients within the ice condenser.

   The CSB will determine from the safety analysis report that the TMD code, without the "augmented flow" correlation, has been utilized to determine the transient pressure response in each subcompartment that contains a high energy line, and in adjoining subcompartments.

   The CSB reviews the maximum calculated differential pressures and pressure profiles for each subcompartment. For plants at the construction permit stage of review, the CSB will ascertain that it is the applicant's intent to design all internal structures with a margin of 40% between the maximum calculated differential pressure and the design differential pressure of the structure or component. At the operating license stage of review, the CSB will ascertain that an appropriate margin exists. However, changes in technology and calculational methods may affect the margin. The CSB will then determine that the maximum calculated differential

pressures do not exceed the design differential pressures for the internal structures. When maximum calculated differential pressures which exceed the pressures used in the design of the internal structures are identified the CSB request the SEB to evaluate the adequacy of the affected internal structures. The loads on components or their supports installed within the compartment due to possible pressure gradients will be evaluated by MEB. The CSB will coordinate the review of dynamic pressure loads for components and equipment supports, and when the design basis loads have been identified the CSB will request the MEB to evaluate the design adequacy of the components and supports.

Modification to the RELAP4 code to include two-phase, two-component mixtures and ice condenser modeling have been made. This will improve the capability of the code for use in short term response analysis of ice condenser plants. The CSB will use the RELAP4 code to conduct confirmatory analyses. The COMPARE code is also being modified to permit the short term response analysis of ice condenser plants.

4. The CSB reviews the methods, input assumptions, and results of the applicant's steam bypass analysis. The applicant's analysis should show considerable margin between the maximum tolerable bypass leakage area and the identifiable bypass area required to allow spray water drainage back to the containment sump. The CSB determines the adequacy of the margins provided for the full spectrum of reactor coolant pipe ruptures. Factors affecting the determination include the proposed inspections and tests to determine bypass leakage area and whether the design of the plant will permit access to seals between the upper and lower compartments for inspection. At the operating license stage, the CSB reviews the proposed technical specifications to assure that adequate surveillance will be maintained for the steam bypass area.

5. The CSB reviews the initial programs for ice loading and subsequent verification of individual ice basket and total ice loads. In addition, it reviews design provisions for monitoring the status of the ice condenser during plant operation to assure that the ice condenser retains its full capability. The CSB also reviews the aspects of the ice condenser design which will allow inspection and functional testing of ice condenser components during various modes of plant operation. Specific areas to be evaluated are the ice condenser temperature instrumentation system, lower inlet door position monitoring system, proposed ice basket inspection programs to determine total ice weight, proposed inspection and testing programs for intermediate and top deck doors, floor drains, lower inlet doors, ice condenser flow passages, divider barrier seals, and access hatches. The CSB determines that the proposed surveillance programs and attendant design provisions fulfill the intent of General Design Criteria 39 and 40. At the operating license stage, the CSB also evaluates the proposed technical specifications that have been established to assure ice condenser operability.

6. The CSB reviews the environmental conditions used in the qualification testing of the return air fan system components. The CSB determines whether the test conditions are representative of post-accident conditions to which the equipment may be subjected. The CSB will ascertain that the equipment can operate in the accident environment for as long as accident conditions require. The CSB reviews analyses demonstrating that where required, the return air fan system and its components are designed to withstand the transient differential pressures to which the systems would be subjected following a loss-of-coolant accident.

   The CSB reviews the provisions made in the design of the return air fan system and the proposed program for periodic inspection and functional testing of the system and components for compliance with the intent of General Design Criteria 39 and 40. The CSB determines the acceptability of the proposed periodic surveillance program for the return air fan system, taking into account the extent and frequency of testing proposed and the practices established for previous ice condenser plants. At the operating license stage, the CSB also evaluates the technical specifications for the return air fan system that have been proposed to assure system operability.

7. The CSB reviews the analysis of the maximum depressurization transient due to inadvertent operation of the containment sprays or return air fans. The CSB reviews the assumed containment initial conditions, methods of calculation, and spray system efficiency to determine whether the containment depressurization analysis is conservative. If the external design pressure of the containment is shown to be exceeded, the CSB will ascertain that containment vacuum relief devices to mitigate the consequences of inadvertent operation of the sprays or fans have been provided, or administrative contrcls have been established and interlocks provided to prevent inadvertent operation of the sprays or fans. If containment vacuum relief devices are used, the CSB reviews the analysis provided to demonstrate that the response time of the relief devices is short enough to prevent depressurization of the containment below the external design pressure. The CSB determines that the vacuum relief devices comply with the requirements of Subsection NE of Section III of the ASME Boiler and Pressure Vessel Code. CSB reviews the design of the vacuum relief devices and proposed inspection and testing programs to ensure that the intent of General Design Criteria 54 and 56 is fulfilled. If administrative controls are established and electrical interlocks provided to preclude inadvertent operation of the sprays or fans, the CSB in conjuction with the ICSB review the acceptability of these provisions from a functional standpoint. At the operating license stage of review, CSB also reviews the proposed technical specifications to assure that adequate surveillance and administrative control will be maintained over the vacuum relief devices.

8. The CSB determines whether instrumentation capable of withstanding post-accident environments, and recording equipment, has been provided to monitor and record the course of an accident within the containment. The CSB also determines that the instrumentation and recording equipment can accomplish the objectives stated in

subsection II. This review effort is coordinated with the ICSB. .The ICSB, in SRP Section 7.3, has review responsibility for the acceptability of, and the qualification test program for the sensing and actuation instrumentation of the plant protection system, the ice condenser temperature monitoring system, and the post-accident monitoring instrumentation and recording equipment.

9. The CSB reviews the minimum containment pressure analysis for the emergency core cooling system performance evaluation in accordance with SRP section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies."

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in SRP Section 6.2.1.

V. REFERENCES

The references for this plan are listed in SRP Section 6.2.1.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.1.1.C PRESSURE-SUPPRESSION TYPE BWR CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Mechanical Engineering Branch (MEB)
Instrumentation and Control Systems Branch (ICSB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

For Mark I, II, and III pressure-suppression type boiling water reactor (BWR) plant containments, the CSB review covers the following areas:

1. The temperature and pressure conditions in the drywell and wetwell due to a spectrum (including break size and location) of postulated loss-of-coolant accidents.

2. The differential pressure across the operating deck of Mark II plants for a spectrum of loss-of-coolant accidents (including break size and location).

3. Suppression pool dynamic effects during a loss-of-coolant accident or following the actuation of one or more reactor coolant system pressure relief valves, including vent clearing, vent interactions, pool swell, pool stratification, and dynamic and asymmetrical loads on suppression pool and other containment structures.

4. The consequences of a loss-of-coolant accident occurring within the containment (wetwell); i.e., outside the drywell (Mark III containments only).

5. The capability of the containment to withstand the effects of steam bypassing the suppression pool.

6. The external pressure capability of the drywell and wetwell, and systems that may be provided to limit external pressures.

7. The effectiveness of static and active heat removal mechanisms.

8. The instrumentation provided to monitor and record containment pressure and temperature and pool water temperature under post-accident conditions.

USNRC STANDARD REVIEW PLAN

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

9. The pressure conditions within subcompartments and acting on system components and supports due to high energy line breaks, e.g., the sacrificial shield structure.

10. The proposed technical specifications, at the operating license stage, pertaining to the surveillance requirements for steam bypass area and vacuum relief devices.

At the request of the CSB, (1) the ICSB will evaluate the functional capability of the post-accident monitoring instrumentation and recording equipment; (2) the MEB will evaluate the acceptability of postulated pipe break sizes and locations and guard pipe designs in SRP Section 3.6.2. The MEB reviews the design of piping and other components for the appropriate combination of pool dynamic loads and other loads in SRP Sections 3.9.2, 3.9.3, and 3.10; and (3) the SEB will evaluate the structural design of unique flow limiting devices used in subcompartments. The SEB also reviews some aspects of guard pipe design in SRP Section 3.8.3.

The evaluation of analytical models used for containment analysis is the responsibility of the CSB.

II. ACCEPTANCE CRITERIA

The following acceptance criteria apply to the design and functional capability of BWR pressure-suppression type containments:

1. For Mark I and II plants at the operating license stage of review, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values. Also, the peak dock differential pressure for Mark II plants should not exceed the design value.

   For Mark III plants, the calculated results for drywell temperature, containment pressure, and differential pressure between the drywell and containment should be based on the General Electric Mark III analytical model (Ref. 30) that was used in the Grand Gulf analysis and evaluated by CSB. The use of this model at the construction permit stage is acceptable if an appropriate margin (see below) between the calculated and design differential pressures is used. The Mark III analytical model will be verified by the large-scale Mark III test results prior to the operating license stage of review for a Mark III plant. If an analytical model other than the General Electric Mark III analytical model identified above is used, the model should be demonstrated to be physically appropriate and conservative to the extent that the General Electric model has been found acceptable. In addition, it will be necessary to demonstrate its performance with suitable test data in a manner similar to that described above.

   Additional analytical efforts are needed to further confirm the Mark III design. These matters were discussed in the ACRS letters issued on December 12, 1974, following its review of the Perry and Allens Creek facilities. The areas of concern

relate to vent interaction, pool stratification, and dynamic and asymmetric loadings. These would also include an evaluation of oscillatory behavior.

For Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15% margin above the peak calculated containment pressure, and the drywell design differential pressure should provide at least a 30% margin above the peak calculated drywell differential pressure.

For Mark III plants at the operating license stage, the peak calculated containment pressure and drywell differential pressure should be less than the design values. In general, it is expected that the peak calculated pressures will be about the same as at the construction permit stage. However, it is possible that the margins may be affected by revised or improved analytical models, test results, or minor changes in the as-built design of the plant.

2. Calculation of dynamic loads on suppression pool retaining structures and structures which may be located directly above the pool, as a result of pool motion during a loss-of-coolant accident or following actuation of one or more primary system pressure relief valves, should be based on appropriate analytical models and supported by applicable test data.

3. High energy lines passing through the containment should be provided with guard pipes or enclosed in other types of protective structures to assure that the suppression pool is not bypassed. If guard pipes are used, they should be designed in accordance with acceptance criteria established by the MEB as set forth in SRP Section 3.6.2.

4. The allowable leakage areas for steam bypass of the suppression pool should be determined for a spectrum of postulated reactor coolant system pipe breaks. The maximum allowable bypass area of the plant should be based on conservative analyses which consider available energy removal mechanisms and the containment design pressure.

5. For Mark I, II and III containments, the maximum allowable leakage area for steam bypass of the suppression pool should be greater than the technical specification limit for leakage measured in periodic drywell-wetwell leakage tests. Specific acceptance criteria for the three types of containments are as follows:

   a. Mark I containments should have a monthly surveillance program along with a testing program to be conducted during each refueling outage for the vacuum breakers.
   b. Mark II containments should be designed to have a steam bypass capability as discussed in Appendix I.

c. Mark III containments should be designed to accommodate, for the spectrum of postulated reactor coolant system pipe breaks reviewed and approved in SRP Section 3.6.2, without exceeding containment design pressure, a minimum bypass leakage area of the order of one square foot. A leakage test of the drywell at about the design pressure should be performed prior to plant operation but as near to startup as feasible. The high pressure test will impose loads on the drywell which are a substantial fraction of the accident loads and will provide the necessary assurance that the drywell, as constructed, conforms to the design bases. Low pressure leakage tests of the drywell should be done periodically thereafter. The acceptance criterion for these tests should be that the measured leakage is less than the leakage corresponding to an equivalent 0.1 ft^2 leakage area in terms of the parameter $A/\sqrt{K}$ at the test pressure, where K is the resistance factor of the actual flow area, A. If the test conditions at a given pressure are representative of a substantial fraction of the loss-of-coolant accident loads, the acceptance criterion becomes 10% of the bypass capability at the given pressure. Test conditions for the drywell should not exceed the conditions assumed for the structural design of the drywell. Structural design of the drywell is evaluated by the SEB in accordance with the acceptance criteria set forth in SRP Section 3.8.3. Testing methods and procedures should be described at the operating license stage of review.

6. For Mark III containments, justification should be provided for any reduction in the containment leak rate claimed for times less than 30 days after a postulated pipe break accident.

7. Provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) of Mark I, II, and III plants, and the operating deck of Mark II plants, against loss of intergrity from negative pressure transients or post-accident atmosphere cooldown:

   a. Structures should be designed to withstand the maximum calculated external pressure.
   b. Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, to assure that the external design pressures of the structures are not exceeded.

   In either case, the external design pressures of the structures, including the design upward deck differential pressure for Mark II plants, should provide an adequate margin above the maximum calculated external pressures to account for uncertainties in the analyses.

8. Instrumentation capable of operating in the post-accident environment should be provided to monitor the drywell and wetwell (or containment) atmosphere pressure and temperature, and pool temperature following an accident. The instrumentation

should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", should be followed.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of BWR pressure-suppression containments. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis for aspects of functional design common to a class of BWR pressure-suppression type containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

1. The CSB reviews the analyses of the drywell and wetwell temperature and pressure response for Mark I and II containments. The CSB performs confirmatory analyses, when necessary, using the CONTEMPT-LT computer code. Input data for the code, including mass and energy release data, are generally taken from the safety analysis report; however, the AB is currently developing a staff model to calculate mass and energy releases.

   The CSB normally analyzes only the design basis loss-of-coolant accident, which has been found from previous reviews to be the recirculation line break for Mark I and II plants. For Mark III plants, the steam line break has been determined to be the design basis loss-of-coolant accident. However, mass and energy releases from the recirculation line break will be evaluated using various flow correlations.

   The CSB evaluates analyses of both the short-term and long-term pressure and temperature responses of Mark III containment plants. For Mark III plants, the peak containment pressure following a loss-of-coolant accident is independent of the postulated pipe break size. The CSB reviews the containment response analysis presented in the safety analysis report to determine that the acceptance criteria in subsection II have been satisfied.

   The CSB and its consultants have reviewed the General Electric Mark III analytical model and have determined that the code appears to calculate the drywell pressure response in an acceptable manner. The code has been verified by the General Electric Mark III test program.

The CSB verifies from the safety analysis report that the General Electric code has been utilized and that the input assumptions to the code are conservative. If analytical methods other than the General Electric model are used, the CSB, in conjunction with its consultants, will initiate a detailed review of the methods. In this case, the CSB reviews the proposed modeling, analytical methods and assumptions, correlation of results with applicable test data, and comparison with other similar analyses, to determine the acceptability of the proposed model.

The CSB reviews analyses of the drywell response to either a recirculation line rupture or a steam line rupture, as presented in the safety analysis report. The CSB determines from the results of these analyses that the "worst" break has been identified in establishing the drywell design differential pressure as well as the design pressure for subcompartments and equipment supports.

Modifications to the CONTEMP-LT computer code have been made which provide the capability to perform confirmatory analyses of the Mark III drywell pressure response.

2. The CSB reviews analyses of the dynamic loads associated with suppression pool motion during loss-of-coolant accident or following actuation of one or more primary system pressure relief valves. The CSB evaluates the analytical methods, input assumptions, and results and determines the acceptability of the analyses by comparing the analytical results to applicable test data (e.g., from the General Electric large-scale Mark III tests).

3. For Mark III plants, the CSB verifies from the safety analysis report that high energy lines which pass through the containment outside the drywell are provided with guard pipes or enclosed in other types of protective structures. If guard pipes are used, the design must meet the acceptance criteria established in SRP Sections 3.6.2 and 3.8.3. For unguarded lines, the CSB reviews analyses of the consequences of postulated ruptures in these lines. The CSB bases its acceptance of the analyses on the conservatism of the methods and assumptions and on the margin provided to assure against exceeding the design pressure of the containment. If leakage detection and isolation equipment are provided, the CSB evaluates the effectiveness of the detection instrumentation and isolation devices to mitigate the consequences of a pipe rupture. The ICSB reviews the electrical design criteria for these systems.

4. The CSB evaluates analyses of bypass leakage capability. The CSB determines the adequacy of proposed bypass leakage tests and surveillance programs based on the results of previous reviews, operating experience at similar plants, and engineering judgment. At the operating license stage, CSB evaluates the proposed technical specifications pertaining to bypass leakage surveillance. CSB will advise the AAB of the bypass leakage.

5. The CSB evaluates the conservatism of potential depressurization transients. If vacuum relief systems are provided, the CSB verifies from the safety analysis report that the design and operating characteristics of the system meet the requirements of Subsection NE of Section III of the ASME Boiler and Pressure Vessel Code. In evaluating surveillance and test programs for vacuum relief systems, the CSB uses the results of previous reviews and operating experience with similar systems to determine their adequacy. At the operating license stage, the CSB reviews the proposed technical specifications to assure that adequate surveillance and administrative control will be maintained over the vacuum relief devices.

6. The ICSB in SRP Section 7.3 has review responsibility for the acceptability of and the qualification test program for the sensing and actuation instrumentation of the plant post-accident monitoring system.

7. Upon request, the SEB will review the design of unique flow limiting devices which are identified during the CSB review of the containment subcompartments.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this SRP section are presented under SRP Section 6.2.1.

V. REFERENCES

The references for this SRP section are those listed in SRP Section 6.2.1, together with the following:

1a. SRP Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."

Appendix I to SRP 6.2.1.1.C
Containment Systems Branch
Steam Bypass for Mark II Containments

A. Background

This appendix pertains to steam bypass from the drywell to the suppression pool air volume in the Mark II containment design. In a pressure suppression-type containment, steam released from the primary system following a postulated LOCA signal is collected in the containment drywell volume and directed through connecting vents to the suppression pool in the containment wetwell volume and steam is condensed as it enters in the suppression pool. Thus no steam enters the wetwell air volume. The potential exists for steam to bypass the suppression pool by leakage through the vacuum breakers or directly from leak paths in the drywell-to-suppression chamber vent pipes, the diaphragm-wall seal around diaphragm penetrations or cracks in the concrete diaphragm.

The capability for steam bypass for small primary system breaks in the Mark II containment design in considerably less than that found in either the Mark I or the Mark III designs. The Mark III design has a capability of $A/\sqrt{K} = 1\ ft^2$, the Mark I design is of the order of $0.02\ ft^2$. The capability of the Mark II containment is approximately 1/3 that for a Mark I containment and 1/143 that for a Mark III containment. Recognizing the low capability for steam bypass in the Mark II design, we required at the CP stage that several of the later Mark II plants investigate additional means to mitigate the consequences of steam bypass leakage.

This Mark II steam bypass position was developed considering our previous positions for the Mark I and Mark III containment designs on the premise that bypass capability for the Mark II design should either be the same or exceed the capability of the Mark I containment. This can be achieved by upgrading the wetwell spray to an engineered safety feature and requiring automatic actuation of the wetwell spray 10 minutes following a break, consistent with our position for Mark III containments. The overall capability of the Mark II containment to accommodate bypass leakage is improved by making the wetwell spray system automatic. This gives further assurance of maintaining containment integrity in the event of a small break. We base our recommendation on this improvement and not on firm criteria as to precise values of spray actuation time and leakage areas; i.e., our engineering judgement; however, to give guidance on spray capacity we believe that the Mark II containment should be designed to accommodate leakage of the order of $0.05\ ft^2$ $(A/\sqrt{K})$.

To provide assurance that the bypass leakage is not substantially increased over the life of a plant, this position includes requirements for leakage tests. The leakage tests include both periodic low pressure leak tests and a preoperational high pressure leak test. The acceptance criterion for the high and low pressure test is a measured leakage not to exceed 1/10th of capability at the test pressure. This is consistent with the test requirements and acceptance criteria for Mark III containments.

The Mark II bypass capability, including the additional capability resulting from the actuation of the wetwell spray system, is still considerably lower than that for a Mark III containment. Therefore, in addition to the above requirements for leakage test and reliance on the wetwell spray system, this position includes surveillance and instrumentation requirements for the vacuum breakers. This is similar to our requirements for Mark I containment.

8. Position

The system used to quench steam bypassing the suppression pool should be designed such that the steam bypass capability for small breaks is on the order 0.05 ft^2 $(A/\sqrt{K})$ for the Mark II containments. This objective can be satisfied by conformance with the criteria described below. Any proposed alternative criteria must be suitably justified by the applicant and reviewed by the NRC staff.

1. Bypass Capability

The Mark II containment should have a steam bypass capability for small breaks of the order of 0.05 ft^2 $(A/\sqrt{K})$.

a. Containment Wetwell Sprays

The wetwell spray system including the electrical instrumentation and controls, should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability, and other appropriate criteria. The wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell. In addition, the instrumentation and control systems provided to actuate the wetwell spray should be actuated by diverse parameters.

If the existing wetwell spray system is to be used to improve the bypass capability, the consequences of actuation of the wetwell spray system on ECCS function and long-term pool cooling considerations should be evaluated to show that minimum ECCS and pool cooling requirements are met.

b. Transient Bypass Capability Analyses

Transient analyses should be provided to establish the capability for a small break. A normal plant shutdown time of 6 hours should be assumed. The results and bases for the analyses should be provided including the following: the pressure history in the drywell and the wetwell; identification and quantification of the static heat sinks and the condensing heat transfer coefficient; spray capacity, efficiency, coverage, start time and temperature history; identification and quantification of heat sources.

2. Leakage Tests and Surveillance Requirements

a. High Pressure Leak Test

A single preoperational high pressure leakage test should be performed on each Mark II unit. The purpose of this test is to detect leakage in the drywell to suppression chamber vent piping, penetrations, downcomers, vacuum breakers, floor seals, vent seals and the diaphragm. This test should be performed at approximately the peak drywell to wetwell differential pressure following the high pressure structural test of the diaphragm.

b. Low Pressure Leak Tests

Post-operational low pressure leakage test should be performed on each Mark II unit to detect leakage in the drywell to suppression chamber vent piping, penetration downcomers, vacuum breakers, floor seals, vent seals and the diaphragm. This test should be performed at each refueling outage at a differential pressure corresponding to approximately the submergence of the vents.

c. Acceptance Criteria for Leakage Tests

The acceptance criteria for both the high and low pressure leakage tests shall be a measured leakage less than 10 percent of the capability of the containment to accommodate bypass leakage at the test pressure.

d. Surveillance Requirements

A visual inspection should be conducted to detect possible leak paths at each refueling outage. Each vacuum relief valve and associated piping should be checked at this time to determine that it is clear of foreign matter.

3. Vacuum Relief Valve Requirements

a. Position Indicators and Alarms

Redundant position indicators should be placed on all vacuum breakers with redundant indication and an alarm in the control room. The vacuum breaker position indicator system should be designed to provide the plant operators with continuous surveillance of the vacuum breaker position. The indicators should have adequate sensitivity to detect a total valve opening, for all valves, that is less than the bypass capability for a small break. The detectable valve opening should be based on the assumption that the valve opening is evenly divided among all the vacuum breakers.

b. Vacuum Valve Operability Tests

All vacuum breakers should be operability tested at monthly intervals to assure free movement of the valves.

C. Implementation

This position will be applied in the review of all CP and OL applications with Mark II containments.

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SECTION 6.2.1.2 SUBCOMPARTMENT ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The CSB reviews the information presented by the applicant in the safety analysis report concerning the determination of the design differential pressure values for containment subcompartments. A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. A short-term pressure pulse would exist inside a containment subcompartment following a pipe rupture within the volume. This pressure transient produces a pressure differential across the walls of the subcompartment which reaches a maximum value generally within the first second after blowdown begins. The magnitude of the peak value is a function of several parameters, which include blowdown mass and energy release rates, subcompartment volume, vent area, and vent flow behavior. A transient differential pressure response analysis should be provided for each subcompartment or group of subcompartments that meets the above definition.

The CSB review includes the distribution of the mass and energy released into the break compartment, nodalization of subcompartments, subcompartment vent flow behavior, and subcompartment design pressure margins.

The CBS review of the subcompartment model includes the basis for the nodalization within each subcompartment, the initial thermodynamic conditions within each subcompartment, the nature of each vent flow path considered, and the extent of entrainment assumed in the vent flow mixture. The review may also include an analysis of the dynamic characteristics of components, such as doors, blowout panels, or sand plugs, that must open or be removed to provide a vent flow path, and the methods and results of components tests performed to demonstrate the validity of these analyses. The analytical procedure to determine the loss coefficients and inertia terms (L/A, ft^{-1}) for each vent flow path, and to predict the vent mass flow rates, is reviewed. The design pressure chosen for each subcompartment is also reviewed.

USNRC STANDARD REVIEW PLAN

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

MEB and SEB are responsible for reviewing the mechanical and structural design of movable and stationary devices provided for vent flow control in subcompartments.

II. ACCEPTANCE CRITERIA

1. The initial atmospheric conditions within a subcompartment should be selected to maximize the resultant differential pressure. An acceptable model would be to assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from these, the selected values should be justified.

   Another model that is also acceptable, for a restricted class of subcompartments, involves simplifying the air model outlined above. For this model, the initial atmosphere within the subcompartment is modeled as a homogeneous water-steam mixture with an average density equivalent to the dry air model. This approach should be limited to subcompartments that have choked flow within the vents. However, the adequacy of this simplified model for subcompartments having primarily subsonic flow through the vents has not been established.

2. Subcompartment nodalization schemes should be chosen such that there is no substantial pressure gradient within a node, i.e., the nodalization scheme should be verified by a sensitivity study that includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes. The nodalization sensitivity study should include consideration of spatial pressure variation, e.g., pressure variations circumferentially, axially and radially within the subcompartment, for use in calculating the transient forces and moments acting on components.

3. If vent flow paths are used which are not immediately available at the time of pipe rupture, the following criteria apply:

   a. The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.

   b. The validity of the analysis should be supported by experimental data or a testing program should be proposed at the construction permit stage that will support this analysis.

   c. The effects of missiles that may be generated during the transient should be considered in the safety analysis.

4. The vent flow behavior through all flow paths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100% water entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental

data. Currently acceptable vent critical flow correlations are the "frictionless Moody" with a multiplier of 0.6 for water-steam mixtures, and the thermal homogeneous equilibrium model for air-steam-water mixtures.

5. At the construction permit stage, a factor of 1.4 should be applied to the peak differential pressure calculated in a manner found acceptable to the CSB for the subcompartment, structure and the enclosed components, for use in the design of the structure and the component supports. At the operating license stage, the peak calculated differential pressure should not exceed the design pressure. It is expected that the peak calculated differential pressure will not be substantially different from that of the construction permit stage. However, improvements in the analytical models or changes in the as-built subcompartment may affect the available margin.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below are followed for the subcompartment analysis review. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by adopting the results of previous reviews of plants with essentially the same subcompartment and high pressure piping design.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The CSB may perform confirmatory analyses of the blowdown mass and energy profiles within a subcompartment. The analysis is done using the RELAP-4 computer program (see Reference 21 for a description of this code) or the COMPARE computer program (Reference 34). The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in this analysis.

The CSB determines the adequacy of the information in the safety analysis report regarding subcompartment volumes, vent areas, vent resistances, and inertia terms. If a subcompartment must rely on doors, blowout panels, or equivalent devices to increase vent areas, or unique flow limiting devices to control vent flows, the CSB reviews the analysis and testing programs that substantiate their use. The MEB and SEB may also be requested to evaluate the mechanical and structural design of such flow control devices.

The CSB reviews the nodalization of each subcompartment to determine the adequacy of the calculational model. As necessary, CSB performs iterative nodalization studies for subcompartments to confirm that sufficient nodes have been included in the model.

The CSB compares the initial subcompartment air pressure, temperature, and humidity conditions to the criteria given in subsection II, above, to assure that conservative conditions were selected.

The CSB reviews the bases, correlations, and computer codes used to predict subsonic and sonic vent flow behavior and the capability of the code to model compressible and incompressible flow. The bases should include comparisons of the correlations to both experimental data and recognized alternate correlations that have been accepted by the staff.

Using the nodalization of each subcompartment as specified in the safety analysis report, the CSB performs analyses using one of several available computer programs to determine the adequacy of the calculated peak differential pressure. The computer program used will depend upon the subcompartment under review as well as the flow regime. At the present time, the two programs used by the CSB are RELAP-4 (Ref. 21) and COMPARE (Ref. 34).

At the construction permit stage, the CSB will ascertain that the subcompartment design procedures include appropriate margins above the calculated values, as given in Subsection II, above.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in SRP Section 6.2.1.

V. REFERENCES

The references for this SRP Section are those listed in SRP Section 6.2.1.

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SECTION 6.2.1.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The CSB reviews analyses of the mass and energy released to the containment during loss-of-coolant accidents (LOCA) in conjunction with the review of the functional capability of the containment structure. While the CPB has the primary responsibility for the review of mass and energy release analyses, the CSB reviews this area as it relates to containment functional design. The review includes the following areas:

1. The sources of the energy assumed to be released to the containment.
2. The applicant's mass and energy release rate calculations for the initial blowdown phase of the accident.
3. For pressurized water reactor (PWR) plants, because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the core reflood and post-reflood phases of the accident.

II. ACCEPTANCE CRITERIA

The following acceptance criteria apply to the mass and energy release analysis for postulated loss-of-coolant accidents:

1. Sources of Energy

The sources of stored and generated energy that should be considered in analyses of loss-of-coolant accidents include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water.

Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. (Ref. 2). However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA.

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

2. Calculations

In general, calculations of the mass and energy release rates for a loss-of-coolant accident should be done in a manner that conservatively establishes the containment internal design pressure; i.e., maximizes the post-accident containment pressure. The criteria given below for each phase of the accident indicate the conservatism that should exist. These calculations should be done for a spectrum of possible pipe breaks to assure that the worst case has been identified. This spectrum should include hot leg, cold leg (pump suction), and cold leg (pump discharge) pipe breaks with cross-sectional areas up to and including a double-ended pipe break, and longitudinal splits in the largest pipes with break areas equal to the cross-sectional area of the pipe.

a. Initial Blowdown Phase

The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions existing at 102% of full power(Ref. 2).

Mass release rates should be calculated using the Moody model (Ref. 22), or a model that is demonstrated to be equally conservative.

Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Calculations of heat transfer from the secondary coolant to the steam generator tubes for PWR's should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.

For Mark I BWR plants, the mass and energy releases from the rupture of a recirculation line should be based on the General Electric blowdown model(Ref. 30). For Mark II and III plants, mass and energy releases from the rupture of a recirculation line or main steam line should be based on a blowdown model which accounts for the short-term (less than one second) pressure response time of the drywell.

The following computer codes and models are acceptable for calculating mass and energy release rates during the blowdown phase, if they incorporate the foregoing criteria: CRAFT (Ref. 23), SATAN VI (Ref. 24), CE FLASH-4 (Ref. 25), and the GE blowdown model (Ref. 30). Other codes will be acceptable for containment analyses if they are approved by CPB and are determined to be conservative for containment analyses.

b. PWR Core Reflood Phase (Cold Leg Breaks Only)

Following initial blowdown of the reactor coolant system, the water remaining in the reactor vessel should be assumed to be saturated and at the level of the bottom of the active core.

Calculations of the core flooding rate should be based on the emergency core cooling system operating condition that maximizes the containment pressure either during the core reflood phase or the post-reflood phase.

Calculations of liquid entrainment; i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR FLECHT experiments (Ref. 26). Liquid entrainment should be assumed to continue until the water level in the core is 2 feet from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 18-inch core level, a linearly increasing CRF to 0.80 at the 24-inch level, and a constant CRF of 0.80 until the water level is 2 feet from the top of the core. Above this level, a CRF of 0.05 may be used.

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the carryout rate fraction of the increased core inlet water temperature caused by steam quenching.

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

c. PWR Post-Reflood Phase (Cold Leg Breaks Only)

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

Steam quenching should be justified by comparison with applicable experimental data.

The results of post-reflood analytical models should be compared to applicable experimental data.

d. PWR Decay Heat Phase (Cold Leg Breaks Only)

The dissipation of core decay heat should be considered during this phase of the accident.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of the mass and energy release analysis for loss-of-coolant accidents. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The CPB and the CSB compare the sources of energy considered in the loss-of-coolant analysis and the methods and assumptions used to calculate the energy available for release from the various sources with the acceptance criteria listed in Section II, above. The CPB determines the acceptability of the analytical models and the assumptions used to calculate the rates of mass and energy release during the initial blowdown, core reflood, and post-reflood phases of a loss-of-coolant accident. The CSB also compares energy inventories at various times during a loss-of-coolant accident to ensure that the energy from the various sources has been accounted for and has been transferred to the containment on an appropriate time scale. The acceptance of the methods and determination of the degree of conservatism is the responsibility of the CPB. In general, such efforts are accomplished through cooperation between branches.

Mass and energy release data for computer codes that have not been previously reviewed and accepted can be compared to the results of computer codes which have been found acceptable, such as the SATA-VI, CRAFT, and CE FLASH-4 computer codes, and the GE blowdown model. FLOOD 1 or FLOOD 2 computer codes (Refs. 17 and 16) are used, as appropriate, to analyze the mass and energy releases for the PWR core reflood phase. The acceptability of assumptions made in the analyses regarding steam quenching may be determined by comparing the results of the analyses with applicable experimental data.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are listed in Standard Review Plan 6.2.1.

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SECTION 6.2.1.4 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY SYSTEM PIPE RUPTURES

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The CSB reviews analyses of the mass and energy released to the containment during a steam or feedwater line break accident in conjunction with its review of the functional capability of the containment structure. The CSB review includes the following areas:

1. The energy sources that are available for release to the containment.

2. The mass and energy release rate calculations.

II. ACCEPTANCE CRITERIA

In addition to the provisions of General Design Criterion 50, the following acceptance criteria apply to the mass and energy release analysis for postulated PWR secondary system pipe ruptures:

1. Sources of Energy

The sources of energy that should be considered in analyses of steam and feedwater line break accidents include: the stored energy in the affected steam generator metal, including the vessel tubing, feedwater line, and steam line; the stored energy in the water contained within the affected steam generator; the stored energy in the feedwater transferred to the affected steam generator prior to the closure of the isolation valves in the feedwater line; the stored energy in the steam from the unaffected steam generator(s) prior to the closure of the isolation valves in the steam generator crossover lines; and the energy transferred from the primary coolant to the water in the affected steam generator during blowdown.

The steam line break accident should be analyzed for various plant conditions from hot standby to 102% of full power.

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2. Mass and Energy Release Rate Calculations

In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be done in a manner that is conservative from a containment response standpoint; i.e., that maximizes the post-accident containment pressure. The following criteria indicate the degree of conservatism that is desired.

Mass release rates should be calculated using the Moody model (Ref. 22), or a model that is demonstrated to be equally conservative.

Calculations of heat transfer to the water in the affected steam generator should be based on nucleate boiling heat transfer.

Calculations of mass release should consider the water in the affected steam generator and feedwater line, the feedwater transferred to the affected steam generator prior to the closure of the isolation valves in the feedwater lines, the steam in the affected steam generator, and the steam coming from the unaffected steam generator(s) as the secondary system is being depressurized prior to the closure of the isolation valves in the steam generator crossover lines.

If liquid entrainment is calculated for steam line breaks, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steam line breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur, to allow selection of the maximum release case. If no liquid entrainment is calculated, a double-ended rupture of the steam line should be assumed.

The single active failure in the steam generator feedwater line isolation provisions or feedwater pumps that optimizes the mass and energy release to the containment, such that the containment peak pressure is maximized, should be assumed to occur in steam and feedwater line break analyses.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of the mass and energy release analysis of secondary coolant system pipe breaks. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The CSB reviews the secondary coolant system pipe break analysis assumptions to determine whether the "worst" pipe break accident case has been identified by the applicant, and whether the analysis was done in a conservative manner from the standpoint of containment pressure. The CPB reviews the acceptability of the analytical models.

This review may involve coordination between members of the CPB and the CSB on the proposed methods and models used for blowdown analyses. The acceptability of the approach used by the applicant is evaluated based on the acceptance criteria in Section II. The CSB also

reviews analyses of postulated single failures of active components in the secondary system, such as steam and feedwater line isolation valves and feedwater pumps, to determine whether the most severe single failure has been selected which allows mass and energy from the unaffected steam generators and the feedwater system to be transferred to the steam generator blowing down.

If liquid entrainment is calculated in the applicant's steam line break model, the CSB and CPB will determine the validity of the experimental data provided to support the entrainment calculation. CSB and CPB will also ascertain that the effect of steam separators located upstream from the postulated steam line break have been taken into account in the analysis. In addition, the CSB reviews the results of a spectrum of steam line breaks, beginning with the double-ended break and decreasing in area until no entrainment occurs, to be sure that the worst steam line break size has been identified.

The CSB performs confirmatory analyses of the containment pressure response to steam and feedwater line breaks inside the containment using the CONTEMPT-LT computer code.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are listed in Standard Review Plan 6.2.1.

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SECTION 6.2.1.5 MINIMUM CONTAINMENT PRESSURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM PERFORMANCE CAPABILITY STUDIES

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

Following a loss-of-coolant accident in a pressurized water reactor (PWR) plant, the emergency core cooling system (ECCS) will supply water to the reactor vessel to reflood, and thereby cool the reactor core. The core flooding rate is governed by the capability of the ECCS water to displace the steam generated in the reactor vessel during the core reflooding period. For PWR plants, there is a direct dependence of core flooding rate on containment pressure; i.e., the core flooding rate will increase with increasing containment pressure. Therefore, as part of the overall evaluation of ECCS performance, the CSB reviews analyses of the minimum containment pressure that could exist during the period of time until the core is reflooded following a loss-of-coolant accident to confirm the validity of the containment pressure used in ECCS performance capability studies. The CSB reviews the assumptions made regarding the operation of engineered safety feature heat removal systems; the effectiveness of structural heat sinks within the containment to remove energy from the containment atmosphere, and other heat removal processes, such as steam in the containment mixing with ECCS water spilling from the break in the reactor coolant system; and in the case of ice condenser containments, mixing with water from melted ice that drains into the lower containment volume. The review is done for all PWR containment types, i.e., dry, subatmospheric, and ice condenser containments.

The RSB is responsible for determining the acceptability of the mass and energy release data used in the minimum containment pressure analysis (see SRP Section 6.3). This information is derived from the applicant's evaluation of ECCS performance capability in accordance with Appendix K to 10 CFR Part 50.

It should be noted that the minimum containment pressure analysis done in connection with ECCS performance evaluation differs from the containment functional performance analysis, in that the conservatisms and margins are taken in opposite directions in the two cases. Thus, the minimum containment pressure analysis required by the regulations

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for ECCS performance evaluation is not conservative with regard to peak containment pressure in the event of a loss-of-coolant accident and cannot be used to determine the containment design basis.

II. ACCEPTANCE CRITERIA

Paragraph I.D.2 of Appendix K to 10 CFR Part 50 requires that the containment pressure used to evaluate the performance capability of a PWR emergency core cooling system not exceed a pressure calculated conservatively for that purpose.

The guidelines given below indicate the conservatism that analyses of the containment response to loss-of-coolant accidents should have for determining the minimum containment pressure for ECCS performance capability studies:

1. Calculations of the mass and energy released during postulated loss-of-coolant accidents should be based on the requirements of Appendix K to 10 CFR Part 50 (Ref. 2).

2. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," delineates the calculational approach that should be followed to assure a conservative prediction of the minimum containment pressure.

III. REVIEW PROCEDURES

The review procedures described below are followed for the review of the minimum containment pressure analysis. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The CSB reviews the analyses in the safety analysis report of the minimum containment pressure following a loss-of-coolant accident. The RSB confirms the validity of the applicant's mass and energy release data. The CSB evaluates the conservativeness of the assumptions used by the applicant regarding the operation of containment heat removal systems and the effectiveness of structural heat sinks, by comparing the applicant's calculational approach to the method outlined in Branch Technical Position CSB 6-1. In certain cases, the CSB may perform confirmatory containment pressure response analyses using the CONTEMPT-LT computer code. In these cases, containment pressure calculated by the CSB is compared to that used in the applicant's evaluation of the performance capability of the emergency core cooling system, to assure that an appropriately conservative value has been used. The CSB advises the RSB of the acceptability of the containment backpressure used in the ECCS performance evaluation.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in SRP Section 6.2.1.

V. REFERENCES

The reference for this section are listed in SRP Section 6.2.1.

BRANCH TECHNICAL POSITION CSB 6-1

MINIMUM CONTAINMENT PRESSURE MODEL
FOR PWR ECCS PERFORMANCE EVALUATION

A. BACKGROUND

Paragraph I.D.2. of Appendix K to 10 CFR Part 50 (Ref. 1) requires that the containment pressure used to evaluate the performance capability of a pressurized water reactor (PWR) emergency core cooling system (ECCS) does not exceed a pressure calculated conservatively for that purpose. It further requires that the calculation include the effects of operation of all installed pressure-reducing systems and processes. Therefore, the following branch technical position has been developed to provide guidance in the performance of a minimum containment pressure analysis. The approach described below applies only to the ECCS-related containment pressure evaluation and not to the containment functional capability evaluation for postulated design basis accidents.

B. BRANCH TECHNICAL POSITION

1. Input Information for Model

a. Initial Containment Internal Conditions

The minimum containment gas temperature, minimum containment pressure, and maximum humidity that may be encountered under limiting normal operating conditions should be used. Ice condenser plants should use the maximum containment gas temperature.

b. Initial Outside Containment Ambient Conditions

A reasonably low ambient temperature external to the containment should be used.

c. Containment Volume

The maximum net free containment volume should be used. This maximum free volume should be determined from the gross containment volume minus the volumes of internal structures such as walls and floors, structural steel, major equipment, and piping. The individual volume calculations should reflect the uncertainty in the component volumes.

d. Purge Supply and Exhaust Systems

If purge system operation is proposed during the reactor operating modes of startup, power operation, hot standby and hot shutdown, the system lines should be assumed to be initially open.

2. Active Heat Sinks

a. Spray and Fan Cooling Systems

The operation of all engineered safety feature containment heat removal systems operating at maximum heat removal capacity; i.e., with all containment spray trains operating at maximum flow conditions and all emergency fan cooler units operating, should be assumed. In addition, the minimum temperature of the stored water for the spray cooling system and the cooling water supplied to the fan coolers, based on technical specification limits, should be assumed.

Deviations from the foregoing will be accepted if it can be shown that the worst conditions regarding a single active failure, stored water temperature, and cooling water temperature have been selected from the standpoint of the overall ECCS model.

b. Containment Steam Mixing With Spilled ECCS Water

The spillage of subcooled ECCS water into the containment provides an additional heat sink as the subcooled ECCS water mixes with the steam in the containment. The effect of the steam-water mixing should be considered in the containment pressure calculations.

c. Containment Steam Mixing With Water from Ice Melt

The water resulting from ice melting in an ice condenser containment provides an additional heat sink as the subcooled water mixes with the steam while draining from the ice condenser into the lower containment volume. The effect of the steam-water mixing should be considered in the containment pressure calculations.

3. Passive Heat Sinks

a. Identification

The passive heat sinks that should be included in the containment evaluation model should be established by identifying those structures and components within the containment that could influence the pressure response. The kinds of structures and components that should be included are listed in Table 1.

Data on passive heat sinks have been compiled from previous reviews and have been used as a basis for the simplified model outlined below. This model is acceptable for minimum containment pressure analyses for construction permit applications, and until such time (i.e., at the operating license review) that a complete identification of available heat sinks can be made. This simplified approach has also been followed for operating plants by licensees complying with Section 50.46(a)(2) of 10 CFR Part 50. For such cases, and for construction permit reviews, where a detailed listing of heat sinks within the containment often cannot be provided, the following procedure may be used to model the passive heat sinks within the containment:

(1) Use the surface area and thickness of the primary containment steel shell or steel liner and associated anchors and concrete, as appropriate.

(2) Estimate the exposed surface area of other steel heat sinks in accordance with Figure 1 and assume an average thickness of 3/8 inch.

(3) Model the internal concrete structures as a slab with a thickness of one foot and exposed surface of 160,000 ft^2.

The heat sink thermophysical properties that would be acceptable are shown in Table 2.

Applicants should provide a detailed list of passive heat sinks, with appropriate dimensions and properties.

b. Heat Transfer Coefficients

The following conservative condensing heat transfer coefficients for heat transfer to the exposed passive heat sinks during the blowdown and post-blowdown phases of the loss-of-coolant accident should be used (see Figure 2):

(1) During the blowdown phase, assume a linear increase in the condensing transfer coefficient from $h_{initial}$=8 Btu/hr-ft^2-°F, at t = 0, to a peak value four times greater than the maximum calculated condensing heat transfer coefficient at the end of blowdown, using the Tagami correlation (Ref. 2),

$$h_{max} = 72.5 \left[\frac{Q}{Vt_p}\right]^{0.62}$$

where h_{max} = maximum heat transfer coefficient, Btu/hr-ft^2-°F

Q = primary coolant energy, Btu

V = net free containment volume, ft^3

t_p = time interval to end of blowdown, sec.

(2) During the long-term post-blowdown phase of the accident, characterized by low turbulence in the containment atmosphere, assume condensing heat transfer coefficients 1.2 times greater than those predicted by the Uchida data (Ref. 3) and given in Table 3.

(3) During the transition phase of the accident, between the end of blowdown and the long-term post-blowdown phase, a reasonably conservative exponential transition in the condensing heat transfer coefficient should be assumed (See Figure 2).

The calculated condensing heat transfer coefficients based on the above method should be applied to all exposed passive heat sinks, both metal and concrete, and for both painted and unpainted surfaces.

Heat transfer between adjoining materials in passive heat sinks should be based on the assumption of no resistance to heat flow at the material interfaces. An example of this is the containment liner to concrete interface.

(4) Variations from the above guidelines may be found acceptable if the overall ECCS performance evaluation model results in an acceptable peak calculated fuel cladding temperature.

C. REFERENCES

1. 10 CFR §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

2. T. Tagami, "Interim Report on Safety Assessment and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," prepared for the National Reactor Testing Station, February 28, 1966 (unpublished work).

3. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

4. Schmitt, R. C., Bingham, G. E., and Norberg, J. A., "Simulated Design Basis Accident Tests of the Carolinas Virginia Tube Reactor Containment - Final Report," IN-1403, Idaho Nuclear Corporation, December 1970.

TABLE 1

IDENTIFICATION OF CONTAINMENT HEAT SINKS

1. Containment Building (e.g., liner plate and external concrete walls, floor, sump, and linear anchors).

2. Containment Internal Structures (e.g., internal separation walls and floors, refueling pool and fuel transfer pit walls, and shielding walls)

3. Supports (e.g., reactor vessel, steam generator, pumps, tanks, major components, pipe supports, and storage racks)

4. Uninsulated Systems and Components (e.g., cold water systems, heating, ventilation and air conditioning systems, pumps, motors, fan coolers, recombiners, and tanks)

5. Miscellaneous Equipment (e.g., ladders, gratings, electrical cables, trays, and cranes)

TABLE 2

HEAT SINK THERMOPHYSICAL PROPERTIES

Material	Density lb/ft^3	Specific Heat Btu/lb-°F	Thermal Conductivity Btu/hr-ft-°F
Concrete	145	0.156	0.92
Steel	490	0.12	27.0

TABLE 3

UCHIDA HEAT TRANSFER COEFFICIENTS

Mass Ratio (lb air/lb steam)	Heat Transfer Coefficient (Btu/hr-ft^2-F°)	Mass Ratio (lb air/lb steam)	Heat Transfer Coefficient (Btu/hr-ft^2-F°)
50	2	3	29
20	8	2.3	37
18	9	1.8	46
14	10	1.3	63
10	14	0.8	98
7	17	0.5	140
5	21	0.1	280
4	24		

Figure 1
Area of Steel Heat Sinks Inside Containment

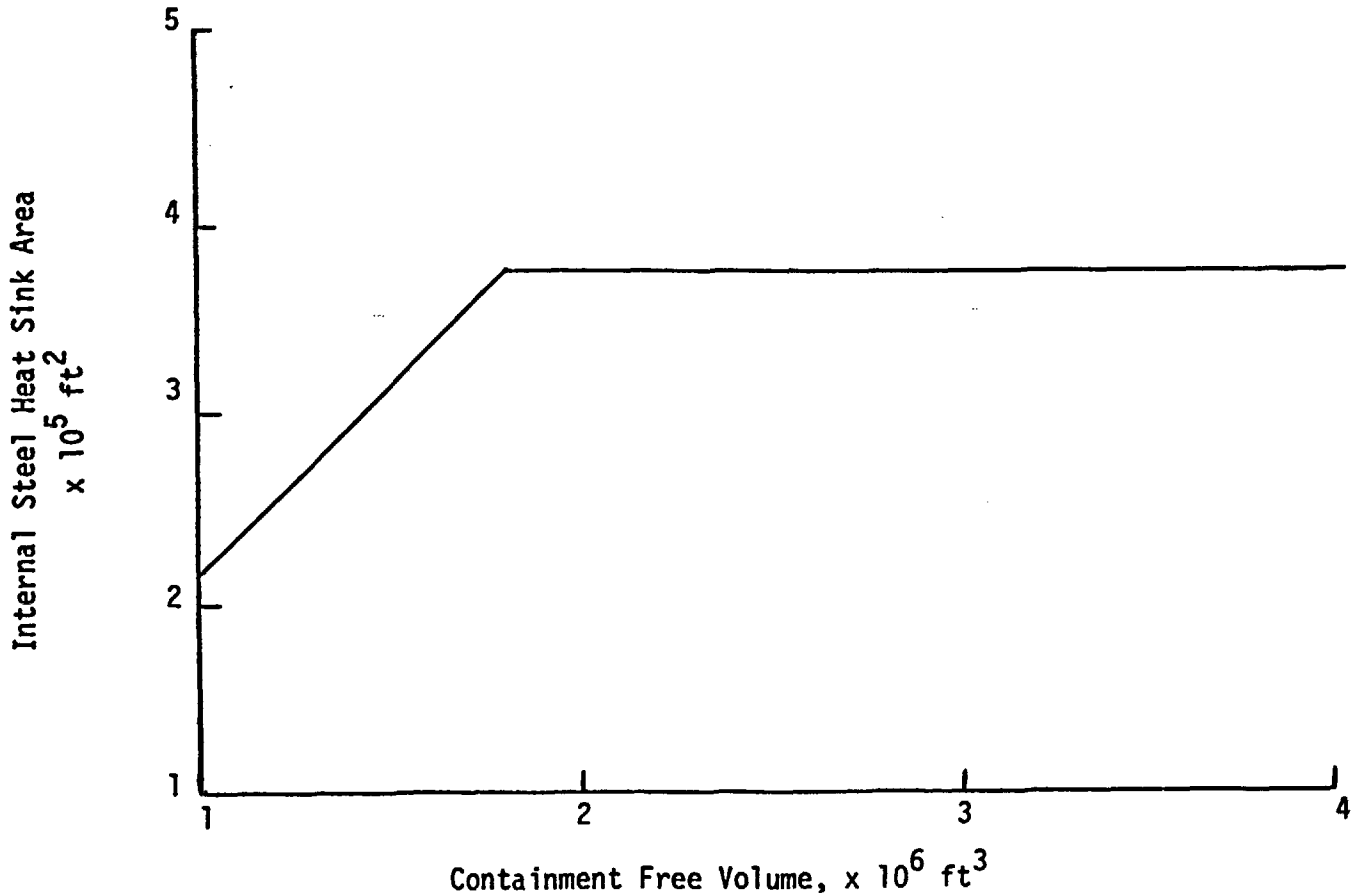
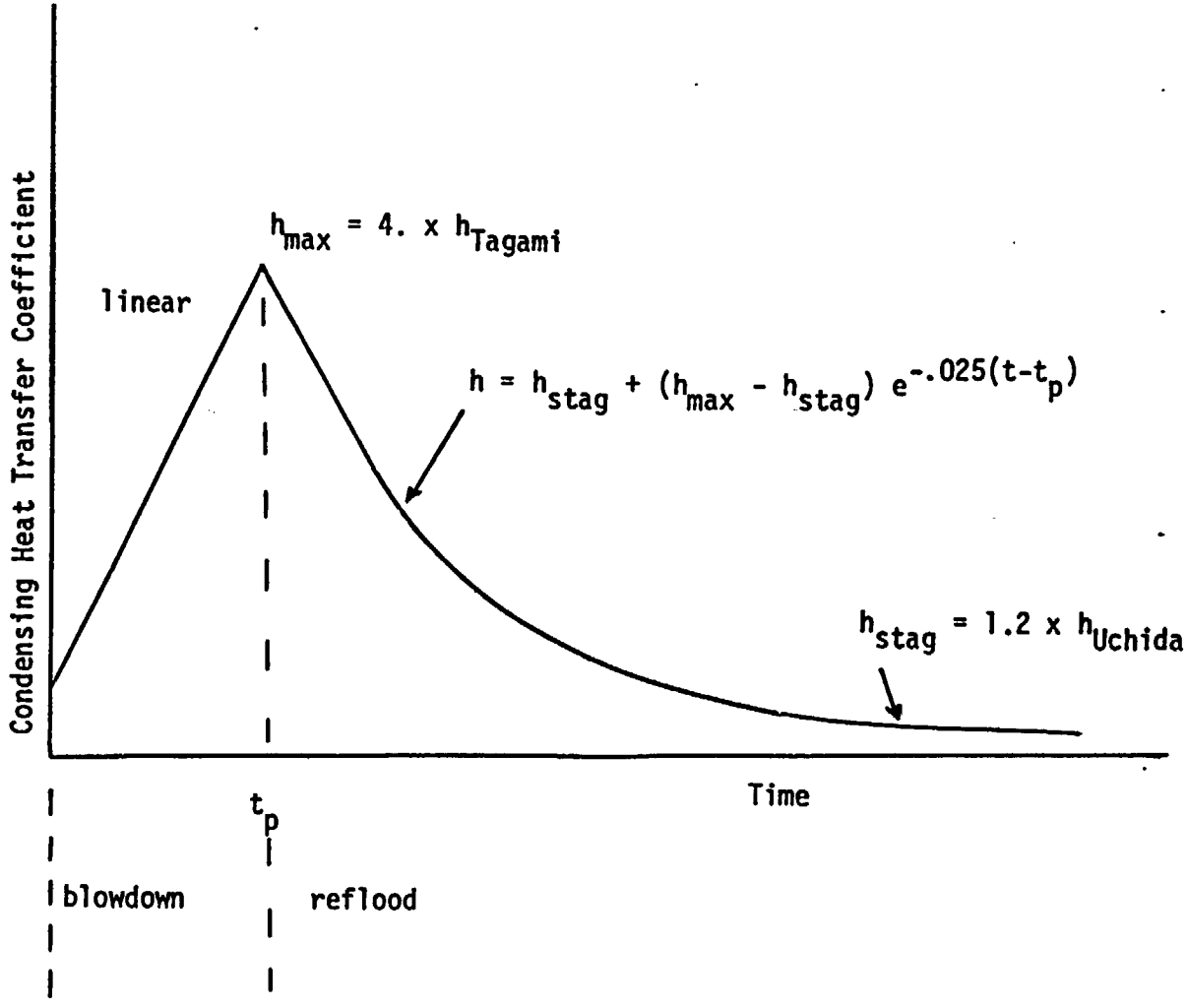


Figure 2

Condensing Heat Transfer Coefficients for Static Heat Sinks



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SECTION 6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Auxiliary Systems Branch (ASB)
Instrumentation and Control Systems Branch (ICSB)
Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

The CSB reviews the information in the applicant's safety analysis report (SAR) concerning containment heat removal under post-accident conditions to assure conformance with the requirements of General Design Criteria 38, 39, 40 and 50. The types of systems provided to remove heat from the containment include fan cooler systems, spray systems, and residual heat removal systems. These systems remove heat from the containment atmosphere and the containment sump water, or the water in the containment wetwell. The CSB review includes the following analyses and aspects of containment heat removal system designs:

1. Analyses of the consequences of single component malfunctions.

2. Analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps.

3. Analyses of the heat removal capability of the spray water system.

4. Analyses of the heat removal capability of fan cooler heat exchangers.

5. The potential for surface fouling of fan cooler, recirculation, and residual heat removal heat exchangers, and the effect on heat exchanger performance.

6. The quality group classification of each system.

7. The seismic design classification of each system.

8. The design provisions and proposed program for periodic inservice inspection and operability testing of each system or component.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

9. The proposed technical specifications for each system.

10. The instrumentation provided to monitor system or component performance.

11. The design of sumps for emergency core cooling and containment spray systems.

12. The effects of debris such as thermal insulation on recirculating fluid systems.

The ASB has the review responsibility for the secondary cooling systems which provide for heat removal from the containment systems to the ultimate heat sink. The ASB is responsible for determining that the systems supplying cooling water to the heat exchangers in the containment heat removal systems meet the design requirements for engineered safety features.

The ICSB has review responsibility for the sensing and actuation instrumentation for the containment heat removal systems (SRP Section 7.3) and for the qualification test programs for their instrumentation.

The AAB reviews fission product control features of containment spray systems (SRP Section 6.5.2).

II. ACCEPTANCE CRITERIA

General Design Criteria 38, 39, 40 and 50 of 10 CFR Part 50, Appendix A, establish requirements for the design, periodic inspection and operability testing, and functional capability of the containment heat removal systems (Refs. 1, 2, 3 and 4). The items listed below amplify these general requirements and form the basis for the staff's detailed review of containment heat removal systems.

1. The containment heat removal systems should meet the redundancy and power source requirements for an engineered safety feature; i.e., the systems should be designed to accommodate a single active failure. The results of failure modes and effects analyses of each system should assure that the system is capable of withstanding a single failure without loss of function. (See Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," for the definition of Single Failure.)

2. The recirculation spray system is required to circulate water in the containment in the long term (after about one hour) following a loss-of-coolant accident, and should be designed to accomplish this without pump cavitation occurring. Therefore, the net positive suction head available to the recirculation pumps should be greater than the required NPSH. A supporting analysis should be presented in sufficient detail to permit the staff to determine the adequacy of the analysis and should show that the available NPSH is greater than the required NPSH. The analysis will be acceptable if it is done in accordance with the guidelines of Regulatory Guide 1.1, i.e., is based on maximum expected temperature of the pumped fluid and with atmospheric pressure in the containment. For clarification, the analysis should be

based on the assumption that the containment pressure equals the vapor pressure of the sump water. This ensures that credit is not taken for containment pressurization during the transient.

The recirculation spray system for a subatmospheric containment is designed to start about five minutes after a loss-of-coolant accident, i.e., during the injection phase of spray system operation. For subatmospheric containments, the guidelines of Regulatory Guide 1.1 as defined above will apply after the injection phase has terminated, which occurs about one hour after the accident. Prior to termination of the injection phase the NPSH analyses should include conservative predictions of the containment atmosphere pressure and sump water temperature transients.

3. Analyses of the heat removal capability of the spray system should be based on the following considerations:

   a. The locations of the spray headers relative to the internal structures.

   b. The arrangement of the spray nozzles on the spray headers and the expected spray pattern.

   c. The type of spray nozzles used and the nozzle atomizing capability, i.e., the spray drop size spectrum and mean drop size emitted from each type of nozzle as a function of differential pressure across the nozzle.

   d. The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.

   The spray systems should be designed to assure that the spray header and nozzle arrangements produce spray patterns which maximize the containment volume covered and minimize the overlapping of the sprays.

4. The design heat removal capability (i.e., heat removal rate vs. containment temperature) of fan coolers should be established on the basis of qualification tests on production units or acceptable analyses that take into account the expected post-accident environmental conditions and variations in major operating parameters such as the containment atmosphere steam-air ratio, condensation on finned surfaces, and cooling water temperature and flow rate. The equipment housing and ducting associated with the fan cooler system should be analyzed to determine that the design is adequate to withstand the effects of containment pressure following a loss-of-coolant accident (see SRP Section 6.2.5). Fan cooler system designs that contain components which do not have a post-accident safety function should be designed such that a failure of nonsafety-related equipment will not prevent the fan cooler system from accomplishing its safety function.

5. The potential for surface fouling of the secondary sides of fan cooler, recirculation, and residual heat removal heat exchangers by the cooling water over the life of the plant and the effect of surface fouling on the heat removal capacity of the heat exchangers should be analyzed and the results discussed in the SAR. The analysis will be acceptable if it is shown that provisions such as closed cooling water systems are provided to prevent surface fouling or surface fouling has been accounted for in establishing the heat removal capability of the heat exchangers.

6. The containment heat removal systems should be designed, fabricated, erected, and tested to Group B quality standards, as recommended by Regulatory Guide 1.26.

7. The containment heat removal systems should be designated Category I (seismic), as recommended by Regulatory Guide 1.29.

8. Provisions should be made in the design of containment heat removal systems for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure-relieving devices, and spray nozzles. The inspection and test program will be acceptable if it is judged by the CSB to be consistent with that proposed for other engineered safety features.

9. Instrumentation should be provided to monitor containment heat removal system and system component performance under normal and accident conditions. The instrumentation should be capable of determining whether a system is performing its intended function, or a system train or component is malfunctioning and should be isolated. The instrumentation should have adequate range, accuracy and response to assure that the parameters can be tracked and recorded. Regulatory Guide 1.97, "Instrumentation For Light Water Cooled Nuclear Power Plants To Assess Plant Conditions During and Following an Accident," should be followed.

10. Provisions should be made to allow drainage of spray and emergency core cooling water to the sumps (recirculation piping suction points). The design of protective screen assemblies around recirculation piping suction points will be acceptable if it is capable of preventing debris from entering the recirculation piping which could impair the performance of system pumps, valves, heat exchangers, or spray nozzles. Regulatory Guide 1.82 (Ref. 8) provides guidance on the design of sumps for emergency core cooling and containment spray systems.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below provide guidance for the review of containment heat removal systems. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a

generic basis for aspects of heat removal systems common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same system.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete. CSB assures that the design and functional capability of the containment heat removal system conform to the requirements of General Design Criteria 38, 39, 40 and 50.

CSB determines the acceptability of a containment heat removal system design by reviewing failure modes and effects analyses of the system to be sure that all potential single failures have been identified and no single failure could incapacitate the entire system; comparing the quality standards applied to the system to Regulatory Guide 1.26; comparing the seismic design classification of the system to Regulatory Guide 1.29; reviewing qualification tests performed on system components such as fan coolers; reviewing the system design provisions for periodic inservice inspection and operability testing to ensure that the system and components are accessible for inspection and all active components can be tested; and reviewing the capability to monitor system performance and control active components from the control room so that the operator can exercise control over system functions or isolate a malfunctioning system component.

For plants at the operating license stage of review, the CSB reviews the proposed technical specifications for containment heat removal systems to assure that limiting conditions for operation and surveillance requirements satisfy the intent of General Design Criteria 39 and 40.

CSB reviews analyses of the net positive suction head available to the recirculation pumps since recirculation system operability is contingent upon adequate NPSH being available to preclude pump cavitation. CSB assures that calculations of the available NPSH are based on transient values of the suction head and the friction head. The CSB determines that the assumption of containment pressure being equal to the vapor pressure of the sump water, has been used in the available NPSH calculations. The CSB reviews information provided by the applicant to identify and justify the conservatisims applied in determining the water level in the containment and the friction losses in the recirculation system suction piping. For example, the uncertainty in determining the free volume in the lower part of the containment that may be occupied by water, and the quantity of water that may be trapped by the reactor cavity and the refueling canal, should be factored into the calculation of the suction head.

The CSB reviews analyses of the available NPSH for subatmospheric containments for the period prior to termination of the injection phase of containment spray to determine that containment pressure and sump water temperature transients have been conservatively used in the NPSH calculations. The CSB reviews information provided by the applicant to identify and justify the conservatisms in the analysis of the containment atmosphere pressure and sump water temperature transients. The CSB also reviews the conservatisms

used in determining the water level in the containment and the friction losses in the recirculation system piping.

The CSB compares the NPSH requirements for the containment heat removal system pumps to the minimum calculated NPSH available to the pumps to assure that a positive margin is maintained. The CSB also reviews the preoperational test programs, and periodic in-service inspection and test programs, to verify that adequate NPSH is available to the pumps and the continuing operability of the pumps during the lifetime of the plant.

If in the judgment of the CSB, the NPSH analyses were not done in a sufficiently conservative manner, confirmatory analyses are performed using the CONTEMPT-LT computer code. See References 10, 11, and 12 for a description of this code.

The CSB also reviews the evaluation of the volume of the containment covered by the sprays and and the extent of overlapping of the sprays with respect to heat removal capabilities. A judgment will be made regarding the acceptability of the spray coverage and extent of overlapping; the volume of the containment covered by the sprays should be maximized and the extent of overlapping kept to a minimum. Elevation and plan drawings of the containment showing the spray patterns are used to determine coverage and overlapping.

In general, the design requirements for the spray systems with respect to spray drop size spectrum and mean drop size, spray drop residence time in the containment atmosphere, containment coverage by the sprays, and extent of overlapping of the sprays are more stringent when the acceptability of the system is being considered from an iodine removal capability standpoint rather than from a heat removal capability standpoint. Consequently, when the iodine removal capability of the system is satisfied, the heat removal capability will be found acceptable. The Accident Analysis Branch is responsible for determining the acceptability of the iodine removal effectiveness of the sprays (See Standard Review Plan 6.5.2). Since all plants do not use the containment sprays as a fission product removal system, the CSB reviews the system for cases where the system is used only as a heat removal system.

CSB reviews analyses of the heat removal capability of the spray system. This capability is a function of the degree of thermal equilibrium attained by the spray water and the volume of the containment covered by the spray water. The spray drop size and residence time in the containment atmosphere determine the degree of thermal equilibrium attained by the spray water. The CSB confirms the validity of the degree of thermal equilibrium attained using the following information: an elevation drawing of the containment showing the locations of the spray headers relative to the internal structures, including fall heights, and the results of the spray nozzle test program to determine the spectrum of drop sizes and mean drop size emitted from the nozzles as a function of pressure drop across the nozzles.

Reference 9 contains information regarding the heating of spray drops in air-steam atmospheres which can be used to determine the validity of the degree of thermal equilibrium of the spray water used in the analyses.

CSB reviews the adequacy of provisions made to prevent overpressurization of fan cooler ducting following a loss-of-coolant accident (Standard Review Plan 6.2.5). CSB reviews the heat removal capability of the fan coolers. The test programs and calculation models used to determine the performance capability of fan coolers are reviewed for acceptability. If the secondary side of a fan cooler heat exchanger is not a closed system, the CSB reviews the potential for surface fouling. The CSB determines whether or not surface fouling impairs the heat removal capability of a fan cooler.

CSB reviews the system provided to allow drainage of containment spray water and emergency core cooling water to the recirculation suction points (sumps). CSB reviews the design of the protective screen assemblies around the suction points. CSB reviews plan and elevation drawings of the protective screen assemblies, showing the relative positions and orientations of the trash bars or grating and the stages of screening, to determine that the potential for debris clogging the screening is minimized. CSB also reviews the drawings to determine that suction points do not share the same screened enclosure. The effectiveness of the protective screen assembly will be determined by comparing the smallest mesh size of screening provided to the clogging potential of pumps, heat exchangers, valves, and spray nozzles. The methods of attachment of the trash bars or grating and the screening to the protective screen assembly structure should be discussed in the SAR and shown on drawings. A discussion of the adequacy of the surface area of screening with respect to assuring a low velocity of approach of the water to minimize the potential for debris in the water being sucked against the screening should be presented. Regulatory Guide 1.82 (Ref. 8) presents guidelines for the acceptability of the design of containment sumps.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"6.2.2 Containment Heat Removal Systems

The containment heat removal systems include (identify the systems).

"The scope of review of the containment heat removal systems for the (plant name) has included system drawings and descriptive information. The review has included the applicant's proposed design bases for the containment heat removal systems, and the analyses of the functional capability of the systems.

"The basis for the staff's acceptance has been the conformance of system designs and design bases to the Commission's Regulations as set forth in the general design criteria, and to applicable regulatory guides, staff technical positions, and

industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of the containment heat removal systems should be discussed.)

"The staff concludes that the design of the containment heat removal systems conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 38, "Containment Heat Removal."

2. 10 CFR Part 50, Appendix A, General Design Criterion 39, "Inspection of Containment Heat Removal System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

4. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

5. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

6. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

7. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

8. Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

9. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems - Part VI, The Heating of Spray Drops In Air-Steam Atmospheres," ORNL-TM-2412, Oak Ridge National Laboratory, January 1970.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)
Auxiliary Systems Branch (ASB)
Analysis Branch (AB)

I. AREAS OF REVIEW

The CSB reviews the information in the applicant's safety analysis report (SAR) concerning the functional capability of the secondary containment system. The secondary containment system includes the outer containment structure of dual containment plants and the associated systems provided to mitigate the radiological consequences of postulated accidents. The secondary containment structure and supporting systems are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. Plant areas and systems contiguous to the secondary containment which also collect and process radioactive material that may leak from the primary containment following an accident are reviewed by the CSB in the same manner as the secondary containment.

The CSB review of the functional capability of the secondary containment system of dual containment designs includes the following points:

1. Analyses of the pressure and temperature response of the secondary containment to a loss-of-coolant accident within the primary containment.

2. Analyses of the effect of openings in the secondary containment on the capability of the depressurization and filtration system to accomplish its design objective of establishing a negative pressure in a prescribed time.

3. Analyses of the pressure and temperature response of the annular region between the primary and secondary containment to a high energy line rupture within the secondary containment.

4. The functional design criteria applied to guard pipes surrounding high energy lines within the secondary containment.

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Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

5. Analyses of any primary containment leakage paths that bypass the secondary containment.

6. The design provisions for periodic leakage testing of secondary containment bypass leakage paths.

7. The proposed technical specifications pertaining to the functional capability of the secondary containment system and the leakage testing of bypass leakage paths.

8. Analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment.

The AAB reviews the design requirements and the periodic inspection and operability test program for the depressurization and filtration systems, from the standpoint of assuring that the systems and system components are functionally capable of depressurizing the secondary containment. The fission product removal capability of the secondary containment supporting systems is reviewed by the AAB under SRP Section 6.5.3.

The AB is responsible for determining the acceptability of the mass and energy release data used in the analysis of the secondary containment pressure response to postulated high energy line breaks, as described in SRP Sections 6.2.1.3 and 6.2.1.4.

The ASB has responsibility for reviewing analyses of pipe ruptures postulated to occur in auxiliary areas or other areas outside primary containment that serve as the secondary containment, i.e., in areas other than the annular region formed by the secondary containment.

II. ACCEPTANCE CRITERIA

1. Analyses of the pressure and temperature response of the secondary containment to a loss-of-coolant accident occurring in the primary containment should be based on the following guidelines:

   a. Heat transfer from the primary to secondary containment should be considered.

      (1) Heat transfer from the primary containment atmosphere to the primary containment structure should be calculated using conservative heat transfer coefficients such as those provided in Branch Technical Position CSB 6-1 (Ref. 4).

      (2) Conductive heat transfer through the primary containment structure and convective heat transfer to the secondary containment atmosphere should be considered.

(3) Radiant heat transfer to the secondary containment should be considered.

b. Adiabatic boundary conditions should be assumed for the surface of the secondary containment structure exposed to the outside environment.

c. The compressive effect of primary containment expansion on the secondary containment atmosphere should be considered.

d. Secondary containment inleakage should be considered.

e. No credit should be taken for secondary containment outleakage.

f. Secondary containment response analyses should be based on the assumption of loss of offsite power and the most severe single active failure in the emergency power system (e.g., a diesel generator failure), in the primary containment heat removal systems, in the core cooling systems, or in the secondary containment depressurization and filtration system. Any delay, due to system design, in actuating the secondary containment depressurization and filtration system should be considered.

g. Heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered.

h. Fan performance characteristics should be considered in evaluating the depressurization of the secondary containment.

2. High energy lines passing through the secondary containment should be provided with guard pipes. Design criteria for guard pipes are given in SRP Section 3.6.2. If guard pipes are not provided, analyses should be provided which demonstrate that both the primary containment structure and the secondary containment structure are capable of withstanding the effects of a high energy pipe rupture occurring inside the secondary containment without loss of integrity.

3. The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment should be specified. Branch Technical Position (BTP) CSB 6-3 (Ref. 3) provides guidance for identifying the leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the secondary containment and other contiguous areas served by ventilation and filtration systems should be described.

4. The negative pressure differential to be maintained in the secondary containment and other contiguous plant areas should be no less than 0.25 inches (water) when compared with adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to assure site boundary exposures less than

those calculated for the design basis accident even if exfiltration occurs. If the leakage rate is in excess of 100% of the volume per day, a special exfiltration analysis should be performed.

5. The secondary containment depressurization and filtration systems should be capable of maintaining a uniform negative pressure throughout the secondary containment, as well as other areas served by the systems.

6. Provisions should be made in the design of the secondary containment system to permit inspection and monitoring of functional capability. The determination of the depressurization time, the secondary containment inleakage rate, the uniformity of negative pressure throughout the secondary containment and other contiguous areas, and the potential for exfiltration should be included in the preoperational and periodic test programs pursuant to General Design Criterion 43.

7. All openings, such as personnel doors and equipment hatches, should be under administrative control. These openings should be provided with position indicators and alarms having readout and alarm capability in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated and confirmatory preoperational tests conducted.

   Some plants may have only portions of the primary containment enclosed, rather than having a secondary containment structure or shield building that completely encloses the primary containment. These enclosed areas are areas into which the primary containment would most likely leak, and they may be equipped with air filtration systems. Quantitative credit cannot be given for the holdup effect of these enclosed areas or for the air filtration systems, to mitigate the radiological consequences of a postulated accident, unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed on a case-by-case basis.

8. The external design pressure of the secondary containment structure should provide an adequate margin above the maximum expected external pressure.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below provide guidance on the review of the secondary containment system. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of secondary containment functional design common to a class of plants, or by adopting the results of previous reviews of similar plants.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

CSB reviews the analytical models used and the assumptions made in the analyses of the pressure and temperature response of the secondary containment to loss-of-coolant accidents in the primary containment. In general, CSB determines that the analyses conservatively predict the secondary containment pressure response. In so doing, CSB compares the analyses to the guidelines in subsection II.

If considered necessary, CSB performs confirmatory analyses of the pressure and temperature response of the secondary containment for loss-of-coolant accidents within the primary containment and for high energy line (e.g., steam line and feedwater line) ruptures occurring within the annular region formed by the secondary containment. The analyses are done using the CONTEMPT-LT computer code (Ref. 2). Upon request by the CSB the AB will evaluate the acceptability of the mass and energy release rates resulting from postulated pipe ruptures under SRP Sections 6.2.1.3 and 6.2.1.4. It should be noted that, for the analysis of the pressure and temperature response in the secondary containment for loss-of-coolant accidents within the primary containment, the present version of the CONTEMPT-LT only has the capability of calculating the pressure in the secondary containment up to the time the depressurization systems are actuated. The code is being improved to permit the calculation of the pressure response for the entire course of an accident.

The analysis will be based on the guidelines given in subsection II, and code input data obtained from the SAR. CSB determines that the secondary containment design pressure is not exceeded and that the depressurization time is consistent with that assumed in the AAB analysis of the radiological consequences of the accident. In addition, CSB determines that the primary containment external design pressure is not exceeded.

CSB determines that all direct leakage paths have been properly identified, and from a review of the proposed leakage testing program that provisions have been made in the design of the plant to measure the fraction of total primary containment leakage that bypasses the secondary containment. The acceptability of the leakage testing program is considered in SRP Section 6.2.6. CSB advises AAB of any inadequacies in the applicant's direct leakage assumptions used in the radiological analysis. At the operating license stage of review, CSB reviews technical specifications which specify the surveillance requirements for leakage testing of the secondary containment bypass leakage paths.

CSB reviews analyses of the capability of the secondary containment system to resist exfiltration under post-accident conditions. If the secondary containment leakage rate is in excess of 100% of the volume per day, CSB advises AAB in order that they may perform a special exfiltration analysis. CSB reviews the preoperational and periodic inservice testing programs to assure that testing will be done to verify the extent of exfiltration.

CSB reviews the proposed secondary containment system testing program and the surveillance requirements in the technical specifications (operating license stage) to assure that tests will be periodically conducted to verify that the prescribed negative pressure can be uniformly maintained throughout the secondary containment. CSB also reviews the testing program and surveillance requirements to assure that tests will be periodically conducted to verify the secondary containment design inleakage rate and to verify the analysis of the depressurization of the secondary containment.

CSB reviews the proposed technical specifications to assure that adequate administrative control will be exercised over the secondary containment openings, such as personnel access doors and equipment hatches. CSB determines from the descriptive information in the SAR that all doors and hatches are provided with position indicators having readout and alarm capability in the main control room. The CSB will ascertain that normally open doors were considered in the analyses of the functional capability of the secondary containment system.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The scope of review of the functional design of the secondary containment system for the has included plan and elevation drawings, system drawings, and descriptive information. This system is provided to control the atmosphere within the secondary containment and contiguous areas. The review has included the applicant's proposed design bases and analyses of the functional capability of the secondary containment system.

"The basis for the staff's acceptance has been the conformance of the functional design and design bases to the Commission's regulations as set forth in the General Design Criteria, and to applicable guides and staff technical positions. (Special problems of exceptions that the staff takes to the design or functional capability of structures or systems should be discussed.)

"The staff concludes that the secondary containment system design conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems."

2. R. J. Wagner and L. L. West, "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear Company, August 1973.

3. Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," attached to this SRP section.

4. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." attached to SRP Section 6.2.1.5.

BRANCH TECHNICAL POSITION CSB 6-3

DETERMINATION OF BYPASS LEAKAGE PATHS IN DUAL CONTAINMENT PLANTS

A. BACKGROUND

The purpose of this branch position is to provide guidance in the determination of that portion of the primary containment leakage that will not be collected and processed by the secondary containment. Bypass leakage is defined as that leakage from the primary containment which can circumvent the secondary containment boundary and escape directly to the environment, i.e., bypasses the leakage collection and filtration systems of the secondary containment. This leakage component must be considered in the radiological analysis of a loss-of-coolant accident.

The secondary containment consists of a structure which completely encloses the primary containment and can be maintained at a pressure lower than atmospheric so that primary containment leakage can be collected or processed before release to the environment. The secondary containment may include an enclosure building which forms an annular volume around the primary containment, the auxiliary building where it completely encloses the primary containment, and other regions of the plant that are provided with leakage collection and filtration systems. Depressurization systems are provided as part of the secondary containment to decrease or maintain the secondary containment volume at a negative pressure.

All primary containment leakage may not be collected because (1) direct primary containment leakage can occur while the secondary containment is being depressurized and (2) primary containment leakage can bypass the secondary containment through containment penetrations and seals which do not terminate in the secondary containment.

Direct leakage from the secondary containment to the environment can occur whenever an outward positive differential pressure exists across the secondary containment boundary. The secondary containment can experience a positive pressure transient following a postulated loss-of-coolant accident in the primary containment as a result of thermal loading and infiltration from the environment and the primary containment that will occur until the depressurization systems become effective. An outward positive differential on the secondary containment wall can also be created by wind loads. In this regard, a "positive" pressure is defined as any pressure greater than -0.25 in. w.g. (water gauge), to account for

wind loads and the uncertainty in the pressure measurements. Whenever the pressure in the secondary containment volume exceeds -0.25 in. w.g., the leakage-prevention function of the secondary containment is assumed to be negated. Since leakage from the secondary containment during positive pressure periods cannot be determined, the conservative assumption is made that, all primary containment leakage is released directly to the environment during these time periods. Therefore, it becomes necessary to determine the time periods during which these threshold conditions exist.

The existence and duration of periods of positive pressure within the secondary containment should be based on analyses of the secondary containment pressure response to postulated loss-of-coolant accidents within the primary containment and the effectiveness of the depressurization systems.

The evaluation of bypass leakage involves both the identification of bypass leakage paths and the determination of leakage rates. Potential bypass leakage paths are formed by penetrations which pass through both the primary and secondary containment boundaries. Penetrations that pass through both the primary and secondary containment may include a number of barriers to leakage (e.g., isolation valves, seals, gaskets, and welded joints). While each of these barriers aid in the reduction of leakage, they do not necessarily eliminate leakage. Therefore, in identifying potential leakage paths, each of these penetrations should be considered, together with the capability to test them for leakage in a manner similar to the containment leakage tests required by Appendix J to 10 CFR Part 50.

B. BRANCH TECHNICAL POSITION

1. A secondary containment structure should completely enclose the primary containment structure, with the exception of those parts of the primary containment that are imbedded in the soil, such as the base mat of the containment structure. For partial dual containment concepts, leak rates less than the design leak rate of the primary containment should not be used in the calculation of the radiological consequences of a loss-of-coolant accident, unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed on a case-by-case basis.

2. Direct leakage from the primary containment to the environment, equivalent to the design leak rate of the primary containment, should be assumed to occur following a postulated loss-of-coolant accident whenever the secondary containment volume is at a "positive" pressure; i.e., a pressure greater than -0.25 in. w.g. Positive pressure periods should

be determined by a pressure response analysis of the secondary containment volume that includes thermal loads from the primary containment and infiltration leakage.

3. The secondary containment depressurization and filtration systems should be designed in accordance with Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants." Preoperational and periodic inservice inspection and test programs should be proposed for these systems and should include means for determining the secondary containment infiltration rate, and the capability of the systems to draw down the secondary containment to the prescribed negative pressure in a prescribed time.

4. For secondary containments with design leakage rates greater than 100 volume percent per day, an exfiltration analysis should be provided.

5. The following leakage barriers in paths which do not terminate within the secondary containment should be considered potential bypass leakage paths around the leakage collection and filtration systems of the secondary containment:

   a. Isolation valves in piping which penetrates both the primary and secondary containment barriers.

   b. Seals and gaskets on penetrations which pass through both the primary and secondary containment barriers.

   c. Welded joints on penetrations (e.g., guard pipes) which pass through both the primary and secondary containment barriers.

6. The total leakage rate for all potential bypass leakage paths, as identified in item 5 above, should be determined in a realistic manner, considering equipment design limitations and test sensitivities. This value should be used in calculating the offsite radiological consequences of postulated loss-of-coolant accidents and in setting technical specification limits with margin for bypass leakage.

7. Provisions should be made to permit preoperational and periodic leakage rate testing in a manner similar to the Type B or C tests of Appendix J to 10 CFR Part 50 for each bypass leakage path listed in item 5 above. An acceptable alternative for local leakage rate testing for welded joints would be to conduct a soap bubble test of the welds concurrently with the integrated (Type A) leakage test of the primary containment required by Appendix J. Any detectable leakage determined in this manner would require repair of the joint.

8. If air or water sealing systems or leakage control systems are proposed to process or eliminate leakage through valves, these systems should be designed, to the extent practical, using the guidelines for leakage control systems given in Regulatory Guide 1.96 (Ref. 3).

9. If a closed system is proposed as a leakage boundary to preclude bypass leakage, then the system should:

   a. Either (1) not directly communicate with the containment atmosphere, or (2) not directly communicate with the environment, following a loss-of-coolant accident.

   b. Be designed in accordance with Quality Group B standards, as defined by Regulatory Guide 1.26. (Systems designed to Quality Group C or D standards that qualify as closed systems to preclude bypass leakage will be considered on a case-by-case basis.)

   c. Meet seismic Category I design requirements.

   d. Be designed to at least the primary containment pressure and temperature design conditions.

   e. Be designed for protection against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features.

   f. Be tested for leakage, unless it can be shown that during normal plant operations the system integrity is maintained.

C. REFERENCES

1. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

2. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

3. Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants."

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)
Instrumentation and Control System Branch (ICSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Reactor Systems Branch (RSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents. This SRP section, therefore, is concerned with the isolation of fluid systems which penetrate the containment boundary, including the design and testing requirements for isolation barriers and actuators. Isolation barriers include valves, closed piping systems, and blind flanges.

The CSB reviews the information presented in the applicant's safety analysis report (SAR) regarding containment isolation provisions to assure conformance with the requirements of General Design Criteria 54, 55, 56 and 57. The CSB review covers the following aspects of containment isolation:

1. The design of containment isolation provisions, including:

   a. The number and location of isolation valves, i.e., the isolation valve arrangements and the physical location of isolation valves with respect to the containment.

   b. The actuation and control features for isolation valves.

   c. The positions of isolation valves for normal plant operating conditions (including shutdown), post-accident conditions, and in the event of valve operator power failures.

   d. The valve actuation signals.

   e. The basis for selection of closure times of isolation valves.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

f. The mechanical redundancy of isolation devices.

g. The acceptability of closed piping systems inside containment as isolation barriers.

2. The protection provided for containment isolation provisions against loss of function from missiles, pipe whip, and earthquakes.

3. The environmental conditions inside and outside the containment that were considered in the design of isolation barriers.

4. The design criteria applied to isolation barriers and piping.

5. The provisions for detecting a possible need to isolate remote-manual-controlled systems, such as engineered safety features systems.

6. The design provisions for and technical specifications pertaining to operability and leakage rate testing of the isolation barriers.

7. The calculation of containment atmosphere released prior to isolation valve closure for lines that provide a direct path to the environs.

PSB has primary responsibility for the qualification test program for electric valve operators, and the ICSB has primary responsibility for the qualification test program for the sensing and actuation instrumentation of the plant protection system located both inside and outside of containment. The MEB has review responsibility for the qualification test program to demonstrate the performance and reliability of containment isolation valves. The MEB and SEB have review responsibility for mechanical and structural design of the containment isolation provisions to ensure adequate protection against missiles, pipe whip, and earthquakes. The AAB reviews the radiological dose consequence analysis for the release of containment atmosphere prior to closure of containment isolation valves in lines that provide a direct path to the environs. The RSB reviews the closure time for containment isolation valves in lines that provide a direct path to the environs, with respect to the prediction of onset of accident induced fuel failure.

II. ACCEPTANCE CRITERIA

The general design criteria establish requirements for isolation barriers in lines penetrating the primary containment boundary. In general, two isolation barriers in series are required to assure that the isolation function is satisfied assuming any single active failure in the containment isolation provisions.

The design of the containment isolation provisions will be acceptable to CSB if the following criteria are satisfied:

1. General Design Criteria 55 and 56 require that lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or

connect directly to the containment atmosphere should be provided with isolation valves as follows:

a. One locked closed isolation valve[1/] inside and one locked closed isolation valve outside containment; or

b. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

c. One locked closed isolation valve inside and one automatic isolation valve[2/] outside containment; or

d. One automatic isolation valve inside and one automatic isolation valve[2/] outside containment.

2. General Design Criterion 57 requires that lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere should be provided with at least one locked closed, remote-manual, or automatic isolation valve[2/] outside containment.

3. The general design criteria permit containment isolation provisions for lines penetrating the primary containment boundary that differ from the explicit requirements of General Design Criteria 55 and 56 if the basis for acceptability is defined. Following are guidelines for acceptable alternate containment isolation provisions for certain classes of lines:

a. Regulatory Guide 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines that are closed both inside and outside containment, are designed to withstand the pressure and temperature conditions following a loss-of-coolant accident, and are designed to withstand dynamic effects, are acceptable without isolation valves.

b. Containment isolation provisions for lines in engineered safety features or engineered safety feature-related systems may include remote-manual valves, but provisions should be made to detect possible leakage from these lines outside containment.

c. Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but provision should be made to detect possible leakage from these lines outside containment.

1/Locked closed isolation valves are defined as sealed closed barriers (see item II.3.f).
2/A simple check valve is not normally an acceptable automatic isolation valve for this application.

d. Containment isolation provisions for lines in the systems identified in items b and c normally consist of one isolation valve inside and one isolation valve outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearest the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

e. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I standards, classified Safety Class 2 (Ref. 5), and should have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak tested, unless it can be shown that the system integrity is being maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in a leak tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

f. Sealed closed barriers may be used in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed isolation valves should be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.

g. Relief valves may be used as isolation valves provided the relief set point is greater than 1.5 times the containment design pressure.

4. Isolation valves outside containment should be located as close to the containment as practical, as required by General Design Criteria 55, 56, and 57.

5. The position of an isolation valve for normal and shutdown plant operating conditions and post-accident conditions depends on the fluid system function. If a fluid system does not have a post-accident function, the isolation valves in the lines should be automatically closed. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. The position of an isolation valve in the event of power failure to the valve operator should be the "safe" position. Normally this position would be the post-accident valve position. All power-operated isolation valves should have position indication in the main control room.

6. There should be diversity in the parameters sensed for the initiation of containment isolation.

7. System lines which provide an open path from the containment to the environs should be equipped with radiation monitors that are capable of isolating these lines upon a high radiation signal. A high radiation signal should not be considered one of the diverse containment isolation parameters.

8. Containment isolation valve closure times should be selected to assure rapid isolation of the containment following postulated accidents. The valve closure time is the time it takes for a power operated valve to be in the fully closed position after the actuator power has reached the operator assembly; it does not include the time to reach actuation signal setpoints or instrument delay times, which should be considered in determining the overall time to close a valve. System design capabilities should be considered in establishing valve closure times. For lines which provide an open path from the containment to the environs; e.g., the containment purge and vent lines, isolation valve closure times on the order of 5 seconds or less may be necessary. The closure times of these valves should be established on the basis of minimizing the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and assure that emergency core cooling system (ECCS) effectiveness is not degraded by a reduction in the containment backpressure. Analyses of the radiological consequences and the effect on the containment backpressure due to the release of containment atmosphere should be provided to justify the selected valve closure time. Additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby and hot shutdown) is provided in Branch Technical Position CSB 6-4 (Ref. 9). For plants under review for operating licenses or plants for which the Safety Evaluation Report for construction permit application was issued prior to July 1, 1975, the methods described in Section B, Items B.1., a, b, d, e, f, and g, B.2 through B.4, and B.5.b, c, and d of Branch Technical Position 6-4 should be implemented. For these plants, BTP Items B.1.c and B.5.a, regarding the size of the purge system used during normal plant operation and the justification by acceptable dose consequence analysis, may be

waived if the applicant commits to limit the use of the purge system to less than 90 hours per year while the plant is in the startup, power, hot standby and hot shutdown modes of operations. This commitment should be incorporated into the Technical Specifications used in the operation of the plant.

9. The use of a closed system inside containment as one of the isolation barriers will be acceptable if the design of the closed system satisfies the following requirements:

   a. The system does not communicate with either the reactor coolant system or the containment atmosphere.

   b. The system is protected against missiles and pipe whip.

   c. The system is designated seismic Category I.

   d. The system is classified Safety Class 2 (Ref. 5).

   e. The system is designed to withstand temperatures at least equal to the containment design temperature.

   f. The system is designed to withstand the external pressure from the containment structural acceptance test.

   g. The system is designed to withstand the loss-of-coolant accident transient and environment.

   Insofar as CSB is concerned with the structural design of containment internal structures and piping systems, the protection of isolation barriers against loss of function from missiles, pipe whip, and earthquakes will be acceptable if isolation barriers are located behind missile barriers, pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment, and the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe shutdown earthquake, as recommended by Regulatory Guide 1.29.

10. The design criteria applied to components performing a containment isolation function, including the isolation barriers and the piping between them, or the piping between the containment and the outermost isolation barrier, are acceptable if:

   a. Group B quality standards, as defined in Regulatory Guide 1.26 are applied to the components, unless the service function dictates that Group A quality standards be applied.

   b. The components are designated seismic Category I, in accordance with Regulatory Guide 1.29.

11. The design of the containment isolation system is acceptable if provisions are made to allow the operator in the main control room to know when to isolate fluid systems that are equipped with remote manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.

12. Provisions should be made in the design of the containment isolation system for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. The acceptance criteria for the leakage rate testing program for containment isolation barriers are presented in SRP section 6.2.6.

For those areas of review identified in subsection 1 of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below provide guidance on review of the containment isolation system. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of containment isolation common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same containment isolation provisions.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection 1. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The CSB determines the acceptability of the containment isolation system by comparing the system design criteria to the design requirements for an engineered safety feature. The quality standards and the seismic design classification of the containment isolation provisions, including the piping penetrating the containment, are compared to Regulatory Guides 1.26 and 1.29, respectively.

The CSB also ascertains that no single fault can prevent isolation of the containment. This is accomplished by reviewing the containment isolation provisions for each line penetrating the containment to determine that two isolation barriers in series are provided, and in conjunction with the PSB by reviewing the power sources to the valve operators.

The CSB reviews the information in the SAR justifying containment isolation provisions which differ from the explicit requirements of General Design Criteria 55, 56 and 57. The CSB judges the acceptability of these containment isolation provisions based on a comparison with the acceptance criteria given in subsection II.

The CSB reviews the position of isolation valves for normal and shutdown plant operating conditions, post-accident conditions, and valve operator power failure conditions as listed in the SAR. The position of an isolation valve for each of the above conditions depends on the system function. In general, power-operated valves in fluid systems which do not have a post-accident safety function should close automatically. In the event of power failure to a valve operator, the valve position should be the position of greater safety, which is normally the post-accident position. However, special cases may arise and these will be considered on an individual basis in determining the acceptability of the prescribed valve positions. The CSB also ascertains from the SAR that all power-operated isolation valves have position indication capability in the main control room.

The CSB reviews the signals obtained from the plant protection system to initiate containment isolation. In general, there should be a diversity of parameters sensed; e.g., abnormal conditions in the reactor coolant system, the secondary coolant system, and the containment, which generate containment isolation signals. Since plant designs differ in this regard and many different combinations of signals from the plant protection system are used to initiate containment isolation, the CSB considers the arrangement proposed on an individual basis in determining the overall acceptability of the containment isolation signals.

The CSB reviews isolation valve closure times. In general, valve closure times should be less than one minute, regardless of valve size. (See the acceptance criteria for valve closure times in subsection II.) Valves in lines that provide a direct path to the environs, e.g., the containment purge and ventilation system lines and main steam lines for direct cycle plants, may have to close in times much shorter than one minute. Closure times for these valves may be dictated by radiological dose analyses or ECCS performance considerations. The CSB will request the AAB or RSB to review analyses justifying valve closure times for these valves as necessary.

The CSB determines the acceptability of the use of closed systems inside containment as isolation barriers by comparing the system designs to the acceptance criteria specified in subsection II.

The MEB and SEB have review responsibility for the structural design of the containment internal structures and piping systems, including restraints, to assure that the containment isolation provisons are adequately protected against missiles, pipe whip, and earthquakes. The CSB determines that for all containment isolation provisions, missile protection and protection against loss of function from pipe whip and earthquakes were design considerations. The CSB reviews the system drawings (which should show the locations of missile barriers relative to the containment isolation provisions) to determine that the isolation provisions are protected from missiles. The CSB also reviews the design criteria applied to the containment isolation provisions to determine that protection against dynamic effects, such as pipe whip and earthquakes, was considered in the design. The CSB will request the MEB to review the design adequacy of piping and [illegible] for which conservative design is assumed to preclude possible breach of system integrity in lieu of providing a leak tight housing.

Systems having a post-accident safety function may have remote-manual isolation valves in the lines penetrating the containment. The CSB reviews the provisions made to detect leakage from these lines outside containment and to allow the operator in the main control room to isolate the system train should leakage occur. Leakage detection provisions may include instrumentation for measuring system flow rates, or the pressure, temperature, radiation, or water level in areas outside the containment such as valve rooms or engineered safeguards areas. The CSB bases its acceptance of the leakage detection provisions described in the SAR on the capability to detect leakage and identify the lines that should be isolated.

The CSB determines that the containment isolation provisions are designed to allow the isolation barriers to be individually leak tested. This information should be tabulated in the safety analysis report to facilitate the CSB review.

The CSB determines from the descriptive information in the SAR that provisions have been made in the design of the containment isolation system to allow periodic operability testing of the power-operated isolation valves and the containment isolation system. At the operating license stage of review, the CSB determines that the content and intent of proposed technical specifications pertaining to operability and leak testing of containment isolation equipment is in agreement with requirements developed by the staff.

IV. EVALUATION FINDINGS

The information provided and the CSB review should support concluding statements similar to the following, to be included in the staff's safety evaluation report:

"The scope of review of the containment isolation system for the (plant name) has included schematic drawings and descriptive information for the isolation provisions for fluid systems which penetrate the containment boundary. The review has also included the applicant's proposed design bases for the containment isolation provisions, and analyses of the functional capability of the containment isolation system.

"The basis for the staff's acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, staff technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to specific containment isolation provisions or the functional capability of the containment isolation system should be discussed.)

"The staff concludes that the containment isolation system design conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

2. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

3. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

4. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

5. Regulatory Guide 1.141, "Containment Isolation Provisions For Fluid Systems."

6. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

7. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

8. Regulatory Guide 1.29, "Seismic Design Classification."

9. Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," attached to this SRP section.

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the purge and vent lines of the containment purge system. It supplements the position taken in SRP section 6.2.4.

While the containment purge system provides plant operational flexibility, its design must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on its use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously.

The containment purge system has been used in a variety of ways, for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. However, the purge and vent lines provide an open path from the containment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR 100 guideline values.

The sizing of the purge and vent lines in most plants has been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on non-routine basis during normal plant operation can be achieved by providing additional purge and vent lines. The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge and vent valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR 100 guideline values. Also, the maximum time for valve closure should not exceed five seconds to assure that the purge and vent valves would be closed before the onset of fuel failures following a LOCA.

The size of the purge and vent lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type; e.g., BWR, PWR or HTGR.

B. BRANCH TECHNICAL POSITION

The system used to purge the containment for the reactor operational modes of power operation, startup, hot standby and hot shutdown; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of cold shutdown and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

   a. The performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in MEB Branch Technical Position MEB-2, Pump and Valve Operability Assurance Program. (Also see SRP Section 3.9.3.) The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the purge line and vent line flows as a function of time up to and during valve closure.

   b. The number of purge and vent lines that may be used should be limited to one purge line and one vent line.

   c. The size of the purge and vent lines should not exceed about eight inches in diameter unless detailed justification for larger line sizes is provided.

d. The containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria.

e. Instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure safety injection actuation, and containment radiation level. If energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can affect the isolation function.

f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds.

g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves, individually, during reactor operation.

5. The following analyses should be performed to justify the containment purge system design:

a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the vent and purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR 100 guideline values.

b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The allowable leak rates of the purge and vent isolation valves should be specified for the spectrum of design basis pressures and flows against which the valves must close.

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

CSB reviews the information presented in the applicant's safety analysis report (SAR) concerning the control of combustible gases in the containment following a loss-of-coolant accident to assure conformance with the requirements of General Design Criteria 41, 42, 43 and 50. Following a loss-of-coolant accident, hydrogen and oxygen may accumulate inside the containment. The major sources of hydrogen and oxygen are: a chemical reaction between the fuel rod cladding and steam, the corrosion of aluminium and other materials by an alkaline spray solution, and the radiolytic decomposition of the water in the reactor core and the containment sump. If excessive hydrogen is generated it may combine with oxygen in the containment atmosphere. For inerted containments, the potential exists for hydrogen to combine with oxygen generated following the accident. The CSB review includes the following general areas:

1. The production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident.

2. The capability to mix the combustible gases with the containment atmosphere and prevent high concentrations of combustible gases in local areas.

3. The capability to monitor combustible gas concentrations within containment.

4. The capability to reduce combustible gas concentrations within containment by suitable means, such as recombination, dilution, or purging.

The CSB review specifically covers the following analyses and aspects of combustible gas control system designs:

1. An analysis of combustible gas (i.e., hydrogen and oxygen) production and accumulation within the containment following a loss-of-coolant accident.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

2. An analysis of the functional capability of the systems provided to mix the combustible gas within the containment.

3. An analysis of the functional capability of the systems provided to reduce combustible gas concentrations within the containment.

4. Analyses of the capability of systems or system components to withstand dynamic effects, such as transient differential pressures that would occur early in the blowdown phase of a loss-of-coolant accident.

5. Analyses of the consequences of single active component malfunctions.

6. The quality classification of each system.

7. The seismic design classification of each system.

8. The results of qualification tests performed on system components to demonstrate functional capability and operability in the accident environment.

9. The design provisions and proposed program (including technical specifications at the operating license stage of review) for periodic inservice inspection, operability testing and leak rate testing of each system or component.

10. The functional aspects of instrumentation provided to monitor system or system component performance.

11. The extent of sharing of system components between sites or between units at a multi-unit site.

Upon request from the CSB, the AAB will evaluate, from a radiological dose standpoint, the acceptability of purge systems provided to control combustible gas concentrations within the containment following a loss-of-coolant accident. In order to compute the purge doses, AAB will need the elapsed time (in days) following a loss-of-coolant accident before purge system operation becomes necessary and the purge rate (in scfm). CSB provides AAB with this information.

At the construction permit (CP) stage of review, the design of the systems provided for monitoring and reducing the concentrations of combustible gases within the containment may not be completely determined. In such cases, CSB reviews the applicant's preliminary designs and statements of intent to comply with the acceptance criteria for such systems. At the operating license (OL) stage, CSB reviews the final designs of these systems to verify that they meet the acceptance criteria detailed below.

II. ACCEPTANCE CRITERIA

1. The analysis of hydrogen and oxygen production in the containment following postulated accidents, for the purpose of establishing the design basis for combustible gas control systems, should be based on the parameters listed in Table 1 of Branch Technical Position CSB 6-2. Branch Technical Position (BTP) CSB 6-2 is an acceptable interim alternative to Regulatory Guide 1.7, pending completion of the rulemaking proceeding on inerting ordered by the Commission in connection with the Vermont Yankee matter, Docket No. 50-271, Memorandum and Order, November 7, 1974, and subsequent revision of Regulatory Guide 1.7. BTP CSB 6-2 supplements and amends Regulatory Guide 1.7 as necessary to take account of the progress in engineered safety features designs and standards since the guide was written and various features of recent containment designs.

2. The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis of the emergency core cooling water and sump water is acceptable if it is equal to or more conservative than the decay energy model given in Branch Technical Position ASB 9-2 in SRP Section 9.2.5.

3. A system should be provided to mix the combustible gases within the containment. The functional design of this system will depend on the type of containment. This system may consist of a fan, a fan cooler, or containment spray. An analysis should be presented which shows that excessive stratification of combustible gases will not occur within the containment or within a containment subcompartment. For containments which rely on convective mixing in conjunction with system operation to mix the combustible gases, the containment internal structures must have design features which promote the free circulation of the atmosphere. An analysis of the effectiveness of these features for convective mixing should be presented. This analysis is acceptable if it can be shown that combustible gases will not accumulate within a compartment or cubicle to form an explosive mixture.

4. The systems provided to reduce the concentration of hydrogen or oxygen in the containment will be accepted, from a functional standpoint, if analyses indicate that a single system train is capable of maintaining the concentration of hydrogen or oxygen below the concentration limits specified in Table 1 of BTP CSB 6-2. Acceptance of the functional capability of the systems is based on confirmatory analyses performed by CSB using system operating parameters presented in the safety analysis report. The proposed operation of the combustible gas control equipment, excluding containment atmosphere dilution (CAD) systems, is acceptable if there is an approprate margin, e.g., on the order of 0.5 v/o, between the limiting hydrogen concentration limit and the hydrogen concentration at which the equipment would be actuated. The proposed operation of CAD systems will be acceptable if there is a margin of 1 v/o between the limiting hydrogen or oxygen concentration limit, depending on which gas is being controlled, and the concentration at which the system would be actuated. This additional margin is needed to allow time for the CAD system to become operational. Repressurization of the containment should be limited to less than 50% of the containment design pressure.

Under loss-of-coolant accident conditions, system components such as ductwork and equipment housings, e.g., for fans, fan coolers, filters, and recombiners, would be subjected to external transient differential pressures and internal pressure surges. These components should be capable of withstanding all related environmental conditions imposed on them, including steam-laden atmosphere differential pressures and pressure surges, without loss of function. A description of the design provisions, such as pressure relief devices or conservative structural design, supporting analyses, and results of tests should be provided to support the conservatism of design.

5. Containment atmosphere sampling or analyzing equipment temperature limitations should be compatible with the temperature of the sample gases.

6. Combustible gas control systems should meet the redundancy and power source requirements for engineered safety features and should be designed to withstand a single active component failure. Supporting failure mode and effects analyses of each system should be provided in the safety analysis report.

7. Combustible gas control systems should be designed, fabricated, erected, and tested to Group B quality standards, as recommended in Regulatory Guide 1.26.

8. Combustible gas control systems, including foundations and supports, should be designated as seismic Category I, i.e., designed to withstand the effects of the safe shutdown earthquake without loss of function, as recommended in Regulatory Guide 1.29.

9. Qualification tests should be performed on system components, such as hydrogen recombiners, combustible gas analyzers, air moving equipment motors, and valve operators. The tests should support the analyses of the functional capability of the equipment and demonstrate that the equipment will remain operable in the accident environment for as long as accident conditions require.

10. Combustible gas control systems should be designed with provisions for periodic inservice inspection, operability testing and leak rate testing of the systems or components. The inspection and test program is acceptable if it is judged to be consistent with that proposed for other engineered safety features.

11. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room.

12. The sharing of system equipment between nuclear power units at a multi-unit site or between sites is acceptable provided (a) the availability of the shared equipment meets the redundancy requirements for an engineered safety feature, (b) the shared equipment is designed to seismic Category I criteria, (c) the shared equipment is

mounted in a seismic Category I structure, (d) adequate design, installation, and procedural provisions have been made, and (e) the stored equipment can be made available to perform its function in a time period that is equal to or less than one-half the time before it is required to operate.

13. Where system equipment is shared between nuclear power units at a multi-unit site or between sites, surveillance programs should be coordinated to assure that redundant equipment is not out of service at the same time.

14. BTP CSB 6-2 recommends that a backup purge system be provided. The backup purge system is not required to be designed to engineered safety features requirements with regard to single failure protection since it is not the primary method for controlling combustible gas concentrations in the containment. The backup purge system is acceptable if purge doses are within the guidelines established in BTP CSB 6-2.

15. If the designs of the combustible gas control systems have not been completed at the construction permit stage of review, they will be acceptable if the preliminary system designs and statements of intent in the SAR conform to BTP CSB 6-2.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures described below provide guidance for the detailed review of the combustible gas control systems. The reviewer selects and emphasizes material from this SRP section, as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of combustible gas control systems design common to a class of plants or by adopting the results of previous reviews of similar plants.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

1. CSB reviews the applicant's analyses of the production and accumulation of oxygen and hydrogen in the containment following postulated loss-of-coolant accidents, to assure that the recommendations and guidelines of BTP CSB 6-2 have been followed. With regard to the extent of metal-water reaction to be considered, the combustible gas control system designs of some boiling water reactor plants with BWR6/Mark III containments have been evaluated and accepted on the basis of an assumed metal-water reaction involving one percent of the cladding mass. Since this assumption is conservative with respect to BTP CSB 6-2 (the BTP would indicate about 0.7% reaction of the cladding mass in these cases), it will continue to be an acceptable basis for these plants, at the option of the applicants. As necessary, the CSB will make confirmatory analyses of combustible gas production and accumulation. These analyses

are done using the COGAP computer code, a description of which is attached as Appendix A to this SRP section. The safety analysis report should contain the required code input data. The purposes of the analyses are:

a. To confirm the predictions of hydrogen and oxygen generation appearing in the safety analysis report.

b. To verify that the systems provided for combustible gas control are capable of maintaining the concentrations of hydrogen and oxygen below the concentration limits specified in Table 1 of BTP CSB 6-2.

c. To confirm the elapsed time before purge system operation becomes necessary.

d. To confirm that the assumed purge rate will maintain combustible gas concentrations within acceptable limits.

The above analyses should be done early in a plant review if this information is needed by AAB to perform the purge dose computations upon which the acceptability of a purge system is based.

2. The combustible gas control systems include systems for mixing the combustible gases, monitoring combustible gas concentrations, and reducing the combustible gas concentrations. In general, all of the combustible gas control systems should meet the design requirements for engineered safety features, as outlined in subsection II. The system description and schematic drawings presented in the safety analysis report should be sufficiently detailed to permit judgments to be made regarding system acceptability.

CSB determines that all potential, single active mechanical failures and passive electrical failures have been identified and that no single failure would incapacitate the entire system. Passive mechanical failures, beyond those possible from missile impact, need not be considered in view of the design and construction standards for the systems.

CSB compares the quality standards applied to the systems to Regulatory Guide 1.26.

CSB compares the seismic design classifications of the systems to Regulatory Guide 1.29.

3. CSB reviews the environmental conditions and duration of tests used for the qualification of system components. CSB determines whether the test conditions and duration are representative of post-accident conditions to which the equipment may be subjected. CSB will ascertain that the equipment can operate in the accident environment for as long as accident conditions require.

4. CSB reviews the provisions made in the design of the systems and the program for periodic inservice inspection and operability testing of the systems or components. The inspections are reviewed with regard to the purpose of each inspection. The operability tests that will be conducted are reviewed with regard to what each test is intended to accomplish. Judgment and experience from previous reviews are used to determine the acceptability of the inspection and test program.

   For plants at the operating license stage of review, CSB reviews the proposed technical specifications for the systems used to control combustible gas concentrations in the containment to assure that the intent of General Design Criteria 41, 42 and 43 are met.

5. CSB reviews the capability to monitor system performance and control active components to be sure that control can be exercised over a system and that a malfunctioning system train or component can be isolated. The instrumentation provided for this purpose should be redundant and should enable the operator to identify the malfunctioning system train or component.

6. CSB reviews the extent of sharing of system equipment between plants at multi-unit sites or between sites to assure that system redundancy requirements are satisfied and that adequate procedural provisions have been made to assure the availability of the shared equipment on a timely basis. The results of CSB analyses of combustible gas production and accumulation are used to confirm the time available following postulated loss-of-coolant accidents to transport the shared equipment to the plant and put it into operation.

7. CSB reviews analyses of the functional capability of the systems provided to mix combustible gases within the containment. CSB reviews the supporting information in the safety analysis report which should include elevation drawings of the containment showing the routing of ductwork and the circulation patterns caused by fans, sprays, or thermal convection. Special attention is paid to interior compartments to assure that combustible gases cannot collect in them without mixing with the bulk containment atmosphere. CSB ensures that interior compartments are identified in the safety analysis report and the provisions made to assure circulation within them are discussed.

   Systems provided to mix the combustible gases within the containment may also be used for containment heat removal, e.g., the fan cooler and spray systems. The acceptability of the design of these systems is considered in the review of the containment heat removal systems in SRP Section 6.2.2.

8. CSB reviews the manner in which the systems provided to reduce combustible gas concentrations will be operated. The concentration at which the system is actuated (the control point) will be determined from the safety analysis report. The margin between the control point and the hydrogen or oxygen concentration limits specified in Table 1 of BTP CSB 6-2 is checked. CSB determines whether the uncertainty in

measuring combustible gas concentrations and the time lag in making the system operational after reaching the control point have been covered by the minimum allowable margin specified in the acceptance criteria.

9. At the construction permit stage of review, the design of the combustible gas control systems may not be complete. In such cases, CSB reviews the preliminary design information and the design criteria that have been established.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The scope of review of the design and functional capability of the combustible gas control systems for the _____ plant has included drawings and descriptive information of the equipment to mix the containment atmosphere, monitor combustible gas concentrations, and reduce combustible gas concentrations within the containment following the design basis accident. The review has also included the applicant's proposed design bases for the combustible gas control systems, and the analyses of the functional capability of the systems provided to support the adequacy of the design bases.

"The basis for the staff's acceptance has been the conformance of system designs and design bases to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, Branch Technical Positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of the combustible gas control systems should be discussed.)

"The staff concludes that the design of the combustible gas control systems conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

2. 10 CFR Part 50, Appendix A, General Design Criterion 42, "Inspection of Containment Cleanup System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup System."

4. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

5. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and the Supplement to Regulatory Guide 1.7.

6. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

7. Regulatory Guide 1.29, "Seismic Design Classification."

8. L. Baker, Jr., and L. C. Just, "Studies of Metal-Water Reaction at High Temperature, III Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, Argonne National Laboratory, May 1962.

9. J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962.

10. H. F. Coward and G. W. Jones, "Limits of Flammability of Gases and Vapors," Bulletin 503, Bureau of Mines (1952).

11. A. O. Allen, "The Radiation Chemistry of Water and Aqueous Solutions," Van Nostrand Co., New York (1961).

12. Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," attached to SRP Section 9.2.5.

13. Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations In Containment Following a Loss-of-Coolant Accident," attached to this SRP section.

APPENDIX A
SRP SECTION 6.2.5
DESCRIPTION OF COGAP

INTRODUCTION

A digital computer program, COGAP (Combustible Gas Analyzer Program), has been developed by the Containment Systems Branch to provide in-house capability for determining hydrogen-oxygen concentrations within reactor containments following loss-of-coolant accidents. The program can also evaluate the performance of a number of combustible control systems. They are the containment atmosphere dilution system (CAD), the recombiner system, and the backup purge system.

DISCUSSION

In the event of a loss-of-coolant accident (LOCA), hydrogen and oxygen gases will be generated within the reactor containment by several reactions. They are:

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant, producing free hydrogen.

2. Radiolytic decomposition of the post-accident emergency cooling solutions, producing both oxygen and hydrogen.

3. Aluminum corrosion by water solutions, producing hydrogen.

4. Zinc corrosion by water solutions, producing hydrogen.

If a sufficient amount of hydrogen is generated, it may react with the O_2 present in the containment atmosphere or, in the case of inerted containments, with the oxygen generated following a LOCA.

The extent of zirconium-water reaction and associated hydrogen production depends strongly upon the course of events assumed for the accident. Analytically the reaction can be described by:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

1 lb Zr → 0.043956 lb H_2

1 lb Zr → 0.021978 lb-mole H_2.

Therefore, one pound of reacted zirconium will produce 0.021978 pound-moles of free hydrogen. Assuming the perfect gas relationship, this is equivalent to 8.4866 scf/lb Zr:

$$V = \frac{MRT}{P}$$

$$V = \frac{0.021978(10.71)(530)}{14.7}$$

$$V = 8.4866 \text{ scf/lb Zr.}$$

The total amount of hydrogen produced is based on the amount of reacted zirconium, as determined by the assumptions given in Branch Technical Position CSB 6-2. The computer program, to maintain a degree of generality, allows the reaction percentage to be specified as an input quantity. The expression used is:

$$WG = (.022)(WZr)(f_{MW})$$

where

WG = pound moles of hydrogen generated

WZr = weight of zirconium fuel element clad

f_{MW} = zirconium-water reaction fraction.

The rate of gas production from radiolysis depends upon the power decay profile and the amount of fission products released to the coolant. The radiolytic hydrogen production rate at time (t) is given by:

$$S_H(t) = \frac{P}{(B)\ (N)} \quad \frac{G_c E_c(t) + G_s E_s(t)}{100}$$

where

$S_H(t)$ = hydrogen production rate, lb-mole/sec

P = operating reactor power level, MWt

B = conversion factor, 454 gm-mole/lb-mole

N = Avogadro's number, 6.023×10^{23} molecules/gm-mole

G_c = radiolytic hydrogen yield in core, molecules/100 ev

$E_c(t)$ = gamma ray fission product energy absorbed by core coolant, ev/sec-MWt

G_s = radiolytic hydrogen yield in solution, $\frac{\text{molecules}}{100 \text{ ev}}$

$E_s(t)$ = energy absorbed in coolant outside core due to fission products dissolved in coolant, ev/sec-MWt.

The quantity $E_c(t)$ is defined by:

$$E_c(t) = (f_\gamma)_c \, H_\gamma(t)$$

where

$(f_\gamma)_c$ = fraction of fission product gamma energy absorbed by coolant in core region

$H_\gamma(t)$ = gamma energy production rate, $\frac{\text{ev}}{\text{sec-MWt}}$.

Similarly, $E_s(t)$ is defined by:

$$E_S(t) = (f_{\gamma+\beta})_s \, H_{\gamma+\beta}(t) + f_I \, H_I(t)$$

Where

$(f_{\gamma+\beta})_s$ = fraction of total solid fission product energy absorbed in coolant outside core

$H_{\gamma+\beta}(t)$ = total solid fission product energy production rate, ev/sec-MWt

f_I = fraction of iodine isotope energy absorbed in coolant outside core

$H_I(t)$ = iodine isotope energy production rate, ev/sec-MWt.

The equations for oxygen generation by radiolysis are identical to those above describing hydrogen evolution except that the yield is one-half that of hydrogen. These equations have been incorporated into the COGAP program. For calculational purposes, the reactor decay profiles ($H_\gamma(t)$, $H_{\gamma+\beta}(t)$, and $H_I(t)$) specified by the ANS-5.1 draft standard for two-year reactor operation have been fitted by several finite exponential series expressions and also incorporated into the program. The resulting equations are:

$$H_\gamma(t) = 10^{22}(5.1912e^{-9.8 \times 10^{-5}t} + 0.8743e^{-6.5 \times 10^{-6}t}$$
$$+ 0.6557e^{-5.7 \times 10^{-7}t} + .4098e^{-7.4 \times 10^{-8}t} + .0150e^{-8.0 \times 10^{-10}t})$$

$$H_{\gamma+\beta}(t) = 2.0 \, H_\gamma(t)$$

$$H_I(t) = 10^{22}(0.8197e^{-6.1 \times 10^{-5}t} + .3279e^{-1.1 \times 10^{5}t}$$
$$+ .0574e^{-1.0 \times 10^{-6}t})$$

where

t = time after reactor shutdown, sec.

Between 400 and 4×10^7 sec, the equations overpredict the standard curve by 20%. The equations underpredict the standard curve soon after shutdown. However, this does not seriously affect the results due to the short time period involved. The equations are equivalent to the afterheat decay curve in BTP ASB 9-2 over the times of interest for post-accident hydrogen generation. It should also be noted that the COGAP formulation overpredicts the radiolytic hydrogen generation by a small amount due to a "double-counting" of the gamma energy of those fission products assumed to be released from the fuel rods.

Hydrogen generation due to aluminimum corrosion is normally considered only when additives are used in the cooling solution. When applicable, gas production is governed by the following expression:

$$S_c(t) = \frac{A\rho BC(t)}{(12)(3.15 \times 10^7)}$$

Where

$S_c(t)$ = hydrogen production rate, lb-mole/sec.

A = surface area of aluminum, ft^2

ρ = aluminum density, lb/ft^3

B = lb-moles of hydrogen per lb of aluminum

C(t) = aluminum corrosion rate, in/year.

The aluminum corrosion rate has been described by an exponential fit in COGAP to account for an increased rate due to high temperatures early in the accident followed by a constant rate for the remaining period of the analysis.

The chemical relationship by which hydrogen is formed has been assumed to be:

$$2\ Al + 3\ H_2O \rightarrow 3\ H_2 + Al_2O_3$$

$$1\ lb\ Al \rightarrow 0.111\ lb\ H_2$$

$$1\ lb\ Al \rightarrow 0.0555\ lb\text{-}mole\ H_2$$

therefore,

$$B = 0.0555\ lb\text{-}mole\ H_2/lb\ Al$$

Zinc corrosion is treated in a similar fashion.

COGAP INPUT REQUIREMENTS

COGAP has been developed to minimize the required input information. All data associated with the power decay profile have been incorporated into the program and need not be entered.

The major input requirements are:

1. Reactor power level, Mwt.
2. Containment free volume, ft^3.
3. Second containment free volume, (wetwell), ft^3.
4. Zirconium cladding weight, pounds.
5. Oxygen dissolved in primary, pound-moles.
6. Hydrogen dissolved in primary, pound-moles.
7. Initial containment pressure, psia.
8. Initial containment temperature, °Rankine.
9. Initial oxygen volume fraction (.209).
10. Recombiner flow rate, cfm.
11. Time recombiner is started, days.
12. Purge rate, cfm.
13. Zirc-water reaction fraction.
14. Aluminum surface area, ft^2.
15. G-H_2, core solution, mole/100 ev.
16. G-H_2, sump solution, mole/100 ev.
17. Fraction of gammas absorbed in coolant in core region.
18. Fraction of solid fission product energy absorbed in solution outside core.
19. Fraction of iodine fission product energy absorbed in solution outside core.

20. Time constant, 9.0×10^8.
21. H_2 concentration fraction at which purging will begin.
22. Time to initiate nitrogen addition, sec.
23. CAD nitrogen flow rate, scf/sec.

BRANCH TECHNICAL POSITION CSB 6-2

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT*

A. BACKGROUND

General Design Criterion 41 requires that systems be provided as necessary to control the concentrations of hydrogen, oxygen, and other substances which may be released into the reactor containment following postulated accidents, to assure that containment integrity is maintained. General Design Criterion 50 requires, in part, that containment be designed to accommodate with margin "metal-water and other chemical reactions that may result from degraded emergency core cooling functioning." This branch technical position (BTP) describes an acceptable method of implementing these criteria for light water reactor plants with cylindrical, zircaloy-clad, oxide fuel. Evaluations of other light water reactor fuels, with stainless steel cladding or with non-cylindrical cladding, will continue to be made on an individual case basis.

Following a loss-of-coolant accident (LOCA), hydrogen gas may accumulate within the containment as a result of:

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.

2. Radiolytic decomposition of the post-accident emergency cooling solutions (oxygen will also evolve in this process).

3. Corrosion of metals by solutions used for emergency cooling or containment spray.

If a sufficient amount of hydrogen is generated, it may react with the oxygen present in the containment atmosphere or, in the case of inerted containments, with the oxygen generated following the accident. The reaction would take place at rates rapid enough to lead to high temperatures and significant overpressurization of the containment, which could result in a leakage rate above that specified in the limiting conditions for operation (technical specifications). Damage to systems and components essential to the continued control of post-LOCA conditions could also occur.

The extent of metal-water reaction and associated hydrogen production depends strongly upon the course of events assumed for the accident and upon the effectiveness of emergency cooling systems. Evaluations of the performance of emergency core cooling systems (ECCS) included as engineered safety features on current light-water-cooled reactor plants have been made by reactor designers using analytical models described in the Commission's Interim Policy Statement of 1971. These calculations are further discussed in the staff's Concluding Statement in the rulemaking hearings, Docket RM-50-1. The result of such evaluations is that for plants of current design, operated in conformance with the Interim Policy Statement, the calculated

*See subsection II. 1 of SRP Section 6.2.5

metal-water reaction amounts to only a fraction of one percent of the fuel cladding mass. As a result of the rulemaking hearing (Docket RM-50-1), the Commission has recently adopted new regulations dealing with the effectiveness of ECCS (10 CFR §50.46).

The staff believes it appropriate to consider the experience obtained from the various ECCS-related analytical studies and test programs such as code developmental efforts, fuel densification, blowdown and core heatup studies, and the PWR and BWR FLECHT tests, and to take account of the foregoing increased conservatism, for plants with ECCS evaluated under §50.46, in setting the amount of initial metal-water reaction to be assumed for the purpose of establishing design requirements for combustible gas control systems. The staff has always separated the design bases for ECCS and for containment systems, and has required containment systems such as the combustible gas control system to be designed to withstand a more degraded condition of the reactor than the ECCS design basis permits. The approach is consistent with provisions of General Design Criterion 50 where the need to provide margins to account for the effects of degraded ECCS function is noted. Although the level of degradation considered might lead to an assumed extent of metal-water reaction in excess of that calculated for acceptable ECCS performance, it does not lead to a situation involving a total failure of the ECCS. The staff feels that this "overlap" in protection requirements provides an appropriate and prudent safety margin against unpredicted events during the course of accidents.

Accordingly, the staff believes that the amount of hydrogen assumed to be generated by metal-water reaction in establishing combustible gas control system performance requirements should be based on the amount calculated in demonstrating compliance with §50.46, but that the amount of hydrogen assumed should include a margin above that calculated. To obtain this margin, the assumed amount of hydrogen should be no less than five times that calculated in accordance with §50.46.

Since the amounts of hydrogen thus determined may be quite small for many plants, as a result of the other more stringent requirements for ECCS performance in the criteria of §50.46, it is consistent with the consideration of the potential for degraded ECCS performance discussed above to establish also a lower limit on the assumed amount of hydrogen generated by metal-water reactions in establishing combustible gas control system requirements. In establishing this lower limit, the staff has noted that the maximum metal-water reaction permitted by the ECCS performance criteria is one percent of the cladding mass.* In fact, the designs of several plants of the BWR6-Mark III type using one percent of the cladding mass as a combustible gas control system basis have recently been reviewed and accepted by the staff and the Advisory Committee on Reactor Safeguards. These plants were reviewed on an individual case basis, since they were the first of the design type. The general and continued use of this "one percent of

*10 CFR Part 50, §50.46(b)(3) "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

the mass" value as a lower limit for assumed hydrogen production, however, would unnecessarily penalize reactors with thicker cladding, since for the same thermal conditions in the core in a postulated LOCA the thicker cladding would not, in fact, lead to increased hydrogen generation. This is because the hydrogen generation from metal-water reaction is a surface phenomenon. The staff considers that a more appropriate basis for setting the lower limit would be an amount of hydrogen assumed to be generated per unit cladding area. It is convenient to specify for this purpose a hypothetical uniform depth of cladding surface reaction. The lower limit of metal-water reaction hydrogen to be assumed is then the hypothetical amount that would be generated if all metal to a specified depth in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) were to react.

In selecting a specified depth to be assumed as a lower limit for all reactor designs, the staff has calculated the depth that could correspond to the "one percent of the mass" value for the current core design with the thinnest cladding. This depth (0.01 times the thickness of the thinnest fuel cladding in use) is 0.00023 inches.

In summary, the amount of hydrogen assumed to be generated by metal-water reaction in determining the performance requirements for combustible gas control systems should be five times the maximum amount calculated in accordance with §50.46, but no less than the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inches.

It should be noted that the extent of initial metal-water reaction calculated for the first core of a plant, and used as a design basis for the hydrogen control system, becomes a limiting condition for all reload cores in that plant unless the hydrogen control system is subsequently modified and reevaluated.

The staff believes that hydrogen control systems in plants receiving operating licenses on the basis of ECCS evaluations under the Interim Policy Statement should continue to be designed for the five percent initial metal-water reaction specified in the original edition of Safety Guide 7. As operating plants are reevaluated as to ECCS performance under 10 CFR §50.46, a change to the new hydrogen control basis enumerated above may be made by appropriate amendments to technical specifications. For plants receiving construction permits on the basis of ECCS evaluations under the Interm Policy Statement, the staff believes that a commitment by the applicant to a specified maximum metal-water reaction, as determined by the provisions of this BTP, is an acceptable alternate basis for the design of a hydrogen control system.

No assumption as to rate of evolution was associated with the magnitude of the assumed metal-water reaction originally given in Safety Guide 7. The metal-water reaction rate is of significance when establishing system performance requirements for containment designs that employ time-dependent hydrogen control features. The staff recognizes that it would be unrealistic to assume an instantaneous release of hydrogen from an assumed metal-water reaction. The staff believes that for the design of a hydrogen control system, it should be assumed that the initial metal-water reaction would occur over a short period of time early in the LOCA transient, i.e., near the end of the blowdown and core refill phases of the LOCA transient.

Any hydrogen thus evolved would mix with steam and air and be rapidly distributed throughout the containment compartments enclosing the reactor primary coolant system by the steam flowing from the postulated pipe break. These compartments include the "drywell" in typical boiling water reactor containments, the "lower volume" of ice condenser containments, and the full volume of "dry" containments. The blowdown and refill phase duration is generally several minutes, and the staff believes that the assumption of a two-minute evolution time at a constant reaction rate, with the resulting hydrogen uniformly distributed in the containment compartments enclosing the primary coolant system, is appropriately conservative for the design of hydrogen control systems. The effects of steam within the containment and containment subcompartments should be considered in the evaluation of the mixture composition.

The rate of production of gases from radiolysis of coolant solutions depends upon (1) the amount and quality of radiation energy absorbed in the specific coolant solutions employed, and (2) the net yield of gases generated from the solutions due to the absorbed radiation energy. Factors such as coolant flow rates and turbulence, chemical additives in the coolant, impurities, and coolant temperature can all exert an influence on the gas yields from radiolysis. The hydrogen production rate from corrosion of materials within the containment, such as aluminum, depends upon the corrosion rate which in turn depends upon such factors as the coolant chemistry, the coolant pH, the metal and coolant temperatures, and the surface area exposed to attack by the coolant. Accurate values of these parameters are difficult to establish with certainty for the conditions expected to prevail following a loss-of-coolant accident.

The staff has reviewed the available information concerning these parameters, including the results of calculations and experiments. Table 1 defines values and other assumptions which the staff believes to be reasonably conservative that may be used for purposes of evaluating the production of combustible gases following a loss-of-coolant accident.

If these assumptions are used to calculate the concentration of hydrogen (and oxygen) within the containment structures of reactor plants following a loss-of-coolant accident, the hydrogen concentration is calculated to reach the flammable limit within periods of less than a day after the accident for the smallest containments and up to more than a month for the largest ones. The hydrogen concentration could be maintained below its lower flammable limit by purging the containment atmosphere to the environs at a controlled rate after the LOCA; however, radioactive materials in the containment would also be released. If purging became necessary shortly after the accident, quantities of such material would be released. The staff believes that the capability for controlled purging should be provided, but that purging should not be the primary means for controlling combustible gases following a LOCA.

The Bureau of Mines has conducted experiments at their facilities with initial hydrogen volume concentrations in the range of four to twelve volume percent. On the basis of these experiments, and of review of reports by others, the staff concludes that a lower flammability limit of four volume percent hydrogen in air or steam-air atmospheres is well established and is adequately conservative. For initial concentrations of hydrogen greater than about six volume percent, it is possible in the presence of sufficient ignition sources that the total accumulated hydrogen could burn in the containment. For hydrogen concentrations in the range of four to six volume percent, partial burning of the excess hydrogen above four volume percent

may occur. The staff believes that a limit of six volume percent would not result in effects that would be adverse to containment systems. Applicants or licensees should demonstrate through supporting analyses and experimental data that containment features and safety equipment required to operate following a LOCA would not be made inoperative by burning of the excess hydrogen, if a design limit in the range of four to six volume percent hydrogen is proposed.

In small containments, the amount of metal-water reaction postulated in Table 1 may result in hydrogen concentrations above acceptable limits. The evolution rate of hydrogen from the metal-water reaction would be greater than that from either radiolysis or corrosion, and since it is difficult for a hydrogen control system to process large volumes of hydrogen very rapidly, an alternative approach is to operate some of the smaller containments with inert (oxygen deficient) atmospheres. This measure, the so-called "inerting" of a containment, provides sufficient time for combustible gas control systems to reduce the concentration of hydrogen following a loss-of-coolant accident before the oxygen generated by radiolysis results in flammable mixtures in the containment. Any requirement for inerting of a containment should be considered on an individual case basis, taking into account the features of the plant, the details of the inservice inspection program for components inside containment, and the need for protection against possible effects from combustible gases.

For all containments, it is advisable to provide means for mixing, sampling, and control of combustible gases resulting from the postulated metal-water reaction, radiolysis, and corrosion following a LOCA, which do not involve releases of radioactive materials to the environment. It is also advisable, as a backup measure, to provide the capability of purging the containment. Filters should be provided as needed in the purge stream to limit the potential release of radioactive iodine and other radioactive materials so that the calculated radiological consequences of the LOCA, including the purge, do not exceed the guideline doses given in 10 CFR Part 100.

Since any system for combustible gas control is designed for the protection of the public in the event of an accident, it should meet the design and construction standards of engineered safety features. Care should be taken in its design to assure that the system itself does not introduce safety problems that may affect containment integrity; for example, if a flame recombiner is used, propagation of flame into the containment should be prevented.

For most reactor plants, operation of the hydrogen control system would not be required for time periods of the order of seven days or more following a postulated design basis LOCA. Thus, it is reasonable that hydrogen control systems need not necessarily be installed at each reactor. Provision for either onsite or offsite storage or a shared arrangement between licensees of plants in close proximity to each other may be developed. An example of an acceptable arrangement would be to provide at least one hydrogen control system per site with the provision that a redundant unit would be available from a nearby site.

B. BRANCH TECHNICAL POSITION

1. All water-cooled power reactor facilities should have the capability for measurement of the hydrogen concentration, for mixing the atmosphere in the containment, and for

controlling combustible gas concentrations without reliance on purging of the containment atmosphere following a loss-of-coolant accident.

2. The continuous presence of combustible gas control equipment at the site may not be necessary provided it is available on an appropriate time scale; however, appropriate design and procedural provisions should be made for its use. In addition, centralized storage facilities that would serve multiple sites may be used provided that these facilities include provisions such as maintenance, protective features, testing, and transportation for redundant units to a particular site.

3. Combustible gas control systems and the provisions for mixing, measuring, and sampling should meet the design, quality assurance, redundancy, energy source, and instrumentation requirements for an engineered safety feature, and the system itself should not introduce safety problems that may affect containment integrity. The combustible gas control system should be designated seismic Category I (see Regulatory Guide 1.29), and the Group B quality standards of Regulatory Guide 1.26 should be applied.

4. All water-cooled power reactors should also have the installed capability for a controlled purge of the containment atmosphere. The purge system need not be redundant nor be designated seismic Category I, except insofar as portions of the system constitute part of the primary containment boundary. Filtration of the purge stream should be provided as necessary to reduce the sum of the long-term doses from the LOCA and the purge to values less than the guidelines of 10 CFR Part 100 at the low population zone outer boundary.

5. The parameter values listed in Table 1 should be used for the purpose of calculating hydrogen and oxygen gas concentrations in containments and evaluating designs provided to control and to purge combustible gases evolved in the course of loss-of-coolant accidents. These values may be changed on the basis of additional experimental evidence and analyses.

6. Materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practical.

7. For plants for which a notice of hearing on the application for a construction permit was published after November 5, 1970:

   a. Plants receiving operating licenses on the basis (in part) of ECCS evaluations under §50.46 should conform to items 1-6, above, prior to operation.

   b. Plants receiving operating licenses on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, should conform, prior to operation, to items 1-6, above, but with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis.

Reevaluations of combustible gas control measures for plants in this category to take account of the change in amount of assumed metal-water reaction may be made at the option of applicants and licensees after submission of §50.46 ECCS analyses and final approval by the staff.

c. Designs of plants receiving construction permits on the basis (in part) of ECCS evaluations under §50.46 should include combustible gas control measures in conformance with items 1-6, above.

d. Designs of plants receiving construction permits on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, should include combustible gas control measures that conform, at the option of applicants, to one of the following:

(1) Items 1-6, above, based on a commitment to a specified maximum metal-water reaction to be calculated according to §50.46.

(2) Items 1-6, above, but with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis.

8. For plants for which a notice of hearing on the application for a construction permit was published between December 22, 1968 and November 5, 1970:

a. A redundant combustible gas control system (such as a recombiner system) as described in items 1 and 2, above, or a repressurization system[1/] designed with redundant elements and designated seismic Category I should be provided unless purging doses are less than the limits given in subparagraph b, below. Purging capability should also be provided as a backup measure to a combustible gas control system, but in this case no purging dose computations need be submitted and the purging system need not have redundant elements or be designated seismic Category I, except insofar as portions of the system constitute part of the primary containment boundary.

b. If the incremental long-term doses from purging in the event of a postulated LOCA are calculated to be less than 2.5 rem whole body and 30 rem thyroid at all points beyond the exclusion area boundary, no combustible gas control systems other than the purging system need be provided. The combination of the dose from the purge and the long-term dose from a postulated LOCA should be below the guidelines of 10 CFR Part 100 at the low population zone outer boundary. Any

[1/] Provisions such as a containment atmospheric dilution system that introduces additional gas into the drywell of some BWR plants may be provided to delay the time to purge on plants in this category; however, the containment should not be repressurized beyond 50% of the containment design pressure.

filtration system for which credit is taken in calculating the purging dose should be redundant, should be designated seismic Category I, and the Group B quality standards of Regulatory Guide 1.26 should be applied. Such filtration systems should be designed, constructed, and tested to meet the recommendations of Regulatory Guide 1.52 to the extent practical. The purging system should be designed so that it is not made inoperative by the failure of any single active component (such as a valve, blower, or electrical power source).

c. For plants receiving operating licenses on the basis (in part) of ECCS evaluations under §50.46, the parameter values listed in Table 1 should be used to calculate combustible gas concentrations in containments and to evaluate designs provided to control and to purge these gases.

For operating plants, or plants receiving operating licenses on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, the parameter values of Table 1 should be similarly used, with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis. Reevaluations of combustible gas control measures for plants in this category to take account of the change in amount of assumed metal-water reaction may be made at the option of applicants and licensees after submission of §50.46 ECCS analyses and final approval by the staff.

d. Combustible gas control systems conforming to this section (B.8) should be provided prior to operation or as soon thereafter as practical.

9. For plants for which a notice of hearing on the application for a construction permit was published before December 22, 1968:

a. Information regarding the calculated dose from purging should be furnished to the staff. If the sum of the long-term doses from a postulated LOCA and the purging dose is below the guidelines of 10 CFR Part 100 at the low population zone outer boundary, no combustible gas control systems other than the purging system need be provided.

b. Any filtration system for which credit is taken in calculating the purging dose should be redundant and designated seismic Category I, and the Group B quality standards of Regulatory Guide 1.26 should be applied. Such filtration systems should be designed, constructed, and tested to meet the recommendations of Regulatory Guide 1.52 to the extent practical.

c. The purging system should be designed so that it is not made inoperative by the failure of any single active component (such as a valve, blower, or electrical power source).

d. If the long-term dose limit of subparagraph a. cannot be met by a purging system with filtration, either a redundant combustible gas control system (such as a recombiner system) as described in items 1 and 2, above, or a repressurization system[1/] with redundant elements and designated seismic Category I should be provided. Purging capability should also be provided as a backup measure for the combustible gas control system, but the purging system need not have redundant filters, be designated seismic Category I, except insofar as portions of the system constitute part of the primary containment boundary, or meet the single failure or long-term dose limit criteria, above.

e. For plants receiving operating licenses on the basis (in part) of ECCS evaluations under §50.46, the parameter values listed in Table 1 should be used to calculate combustible gas concentrations in containments and to evaluate designs provided to control and to purge these gases.

For operating plants, or plants receiving operating licenses on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, the parameter values of Table 1 should be similarly used, with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis. Reevaluations of combustible gas control measures for plants in this category to take account of the change in amount of assumed metal-water reaction may be made at the option of applicants and licensees after submission of §50.46 ECCS analyses and final approval by the staff.

f. Schedules for installation of purging systems or other combustible gas control systems should be considered on an individual case basis.

C. REFERENCES

The references for this branch technical position are the same as those for SRP Section 6.2.5, given in subsection V.

[1/] Ibid., page 21.

TABLE 1

1.	Fraction of fission product radiation energy absorbed by the coolant[1/]	a. Beta (1) Betas from fission products in the fuel rods: 0 (2) Betas from fission products intimately mixed with coolant: 1.0 b. Gamma (1) Gammas from fission products in the fuel rods, coolant in core region: 0.1[2/] (2) Gammas from fission products intimately mixed with coolant, all coolant: 1.0
2.	$G(H_2)$[1/]	0.5 molecules/100 ev
3.	$G(O_2)$[1/]	0.25 molecules/100 ev
4.	Extent and evolution time of initial core metal-water reaction hydrogen production from the surrounding fuel.	Hydrogen production is five times the amount from the maximum calculated reaction under 10 CFR §50.46, or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.00023 inches, whichever is greater, in two minutes.
5.	Aluminum corrosion rate for aluminum exposed to alkaline solutions.	200 mils/yr (This value should be adjusted upward for higher temperatures early in the accident sequence)
6.	Fission product distribution model.	a. 50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water. b. All noble gases are released to the containment. c. All other fission products remain in fuel rods.
7.	a. Hydrogen concentration limit. b. Oxygen concentration limit.	4 volume percent[3/] 5 volume percent (This limit should not be exceeded if more than 6 v/o hydrogen is present.)

FOOTNOTES FOR TABLE 1

1/ For water, borated water, and borated alkaline solutions; for other solutions, data should be presented.

2/ This fraction is thought to be conservative; further analysis may show that it should be revised.

3/ The 4 v/o hydrogen concentration limit should not be exceeded if burning is to be avoided and more than 5 v/o oxygen is present in containment.

This amount may be increased to 6 v/o, with the assumption that the 2 v/o excess hydrogen would burn in the containment (if more than 5 v/o oxygen is present). The effects of the resultant energy and burning should not create conditions exceeding the design conditions of either the containment or safety equipment necessary to mitigate consequences of a LOCA. Applicants and licensees should demonstrate such capability by suitable analyses and qualification test results.

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SECTION 6.2.6 CONTAINMENT LEAKAGE TESTING

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - None

I. AREAS OF REVIEW

Information describing the reactor containment leakage testing program is reviewed by the CSB for conformance to 10 CFR 50 Appendix J and General Design Criteria 52, 53 and 54.

The CSB review of the reactor containment leakage testing program covers the following specific areas:

1. Containment integrated leakage rate tests (Type A tests as defined by Appendix J), including pretest requirements, general test methods, acceptance criteria for pre-operational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria, and scheduling of tests.

2. Containment penetration leakage rate tests (Type B tests as defined by Appendix J), including identification of containment penetrations, general test methods, test pressures, acceptance criteria, and scheduling of tests.

3. Containment isolation valve leakage rate tests (Type C tests as defined by Appendix J), including identification of isolation valves, general test methods, test pressures, acceptance criteria, and scheduling of tests.

4. Technical specifications pertaining to containment leakage rate testing are reviewed at the operating license (OL) stage.

In addition to the tests described above, CSB reviews the special leakage testing programs that may be needed for the secondary containments for plants using the dual containment concept. Dual containments are proposed for some plants because of site limitations. The intent of the dual containment is to collect and process reactor containment leakage. Testing programs to ensure that leakage will be contained as proposed by applicants using this kind of containment are reviewed by CSB (see SRP 6.2.3).

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

II. ACCEPTANCE CRITERIA

The reactor containment leakage rate testing program, as described in the SAR, will be acceptable if it meets the requirements stated in Appendix J to 10 CFR Part 50. Appendix J provides the test requirements and acceptance criteria for preoperational and periodic leak testing of the reactor containment, and of systems and components which penetrate the containment. Exceptions to Appendix J requirements will be reviewed on a case-by-case basis.

Conformance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of the following General Design Criteria applicable to containment leakage rate testing:

(a) General Design Criterion 52, Capability for Containment Leakage Rate Testing;
(b) General Design Criterion 53, Provisions for Containment Testing and Inspection;
(c) General Design Criterion 54, Piping System Penetrating Containment.

The minimum acceptable design containment leakage rate shall not be less than 0.1% per day.

The leakage limits of the secondary containments of dual-type containments are acceptable if they are based on the limits used in the analysis of the secondary containment depressurization time. These tests should be conducted at each refueling or at intervals not exceeding 18 months. The test limits should be consistent with the limits used for direct leakage in the analysis of the radiological consequences by AAB. Potential bypass leak paths (identified in accordance with Branch Technical Position CSB 6-3, Determination of Bypass Leakage Paths in Dual Containment Plants) should be locally leak tested in accordance with the requirements of Appendix J.

Leak testing of instrumentation lines that penetrate containment may be done in conjunction with either the local leak rate tests or the containment integrated leak rate test (CILRT). Instrumentation lines that are not locally leak tested should not be isolated from the containment atmosphere during the performance of the CILRT. The measured leakage rates from instrumentation lines that are locally leak tested should be added to the CILRT result. Provisions should be made to assure that instrumentation lines isolated during the CILRT are restored to their operable status following the test.

Leak testing of hydrogen recombiner systems located outside containment should be included in the local leak rate test program. A local leak test should be done at the time of the CILRT and the measured leak rate added to the CILRT result. Alternately, the recombiner system may be open to the containment atmosphere during the performance of the CILRT.

All leakage tests, performed by either pneumatic or hydrostatic means, should have the capability to quantify the leakage rates either explicitly or by a conservative bounding method.

At the construction permit (CP) stage, the applicant should identify all containment isolation valves that will be locally (Type C) leak tested with the test pressure applied in a direction opposite to that which would occur under accident conditions, and commit to justify at the OL stage, that such testing will result in equivalent or more conservative results.

For leak testing of main steam isolation valves in boiling water reactor plants, a test pressure of less than P_a and the test acceptance criteria should be justified and included in the plant Technical Specifications.

Hydrostatic testing of containment isolation valves is permissible if the line is not a potential containment atmosphere leak path, and may be found acceptable if it can be demonstrated in accordance with the requirements of Section III.C of Appendix J, that a liquid inventory is available to maintain a water seal (while assuming the single failure of any active component) during the post-accident period. Limits for liquid leakage should be assigned to these valves based on analysis and included in the plant Technical Specifications.

The test, vent and drain (TVD) connections that are used to facilitate local leak testing and the performance of the containment integrated leak rate test, should be under administrative control, and should be subject to periodic surveillance, to assure their integrity and verify the effectiveness of administrative controls.

Airlocks should be leak tested prior to initial criticality and at six-month intervals thereafter, at the calculated peak containment internal pressure corresponding to the design basis accident (P_a). Airlocks opened during periods when containment integrity is not required shall be tested at the end of such periods at P_a.

Airlocks opened during periods when containment integrity is required should be tested within three days after being opened. When multiple openings are required, an airlock should be tested at least once every three days during the period of frequent openings. For the case where dual seals are incorporated in the airlock door design, the application of test pressure between the door seals fulfills the three day test requirement.

In the event testing cannot be at P_a, the test pressure should be as stated in Technical Specifications. Airlock door seal testing should not be substituted for the six-month airlock test at P_a. The acceptance criteria for airlock testing should be stated in the plant Technical Specifications.

The testing requirements for BWR drywell steam bypass are discussed in SRP Section 6.2.1.1.C.

III. REVIEW PROCEDURES

At the CP stage, the CSB will review the preliminary design provisions that will permit containment leak testing to be done in accordance with the requirements of Appendix J.

In some instances, however, the applicant may not be able to address specific aspects of the leak testing program because of incomplete designs. Under these circumstances the CSB will review design criteria, and other commitments, that will assure compliance with the requirements of Appendix J. In addition, the CSB will review the applicant's rationale for concluding that the requirements of Appendix J will be met.

At the OL stage, the CSB reviews the containment final design and verifies that the containment leak testing program meets the requirements of Appendix J. In addition, the CSB reviews the plant Technical Specifications for completeness and for conformance to Appendix J.

The review of the reactor containment leakage rate test program at the OL stage specifically includes the following:

1. Containment Integrated Leakage Rate Test (Type A Test)

   Those systems not vented or drained should be identified and the reason for not venting or draining should be stated. Piping and instrumentation diagrams and process flow drawings are used by the reviewer to confirm that in the vented and drained condition, the isolation valves are exposed to the test air pressure and differential pressure, i.e., the systems are vented and drained both upstream and downstream of the containment isolation valves.

2. Containment Penetration Leakage Rate Test (Type B Test)

   All containment penetrations should be listed in the test program. By reference to piping and instrumentation diagrams, the reviewer confirms that all penetrations have been listed. The program should identify any penetration not requiring leakage testing and the reason for not requiring a test should be stated. The reviewer confirms that those penetrations not requiring testing cannot result in leakage to the atmosphere during normal operation or a LOCA.

   Test pressures for containment penetrations should be stated in the test program and in the Technical Specifications. The test pressure is acceptable if it is the maximum calculated containment accident pressure.

3. Containment Isolation Valve Leakage Rate Test (Type C Test)

   All containment isolation valves requiring a Type C test should be listed in the test program. By reference to the piping and instrumentation diagrams, the reviewer confirms that all isolation valves to be tested have been listed.

   Test pressures for isolation valve Type C tests should be included in the test program and Technical Specifications.

   Special testing procedures for dual-type containments should be identified.

CSB assures that the applicant has provided a leakage testing program and has specified the maximum leakage which may occur from bypass (or dilution) leakage for dual-type containments. Potential leakage paths which bypass the annulus or the auxiliary building areas or may leak directly to atmosphere must be identified. The total amount of containment bypass leakage to the environment must be specified and included in the technical specifications. The reviewer determines that the test provisions are adequate to confirm the bypass leakage specified.

Preoperational and periodic test reports are primarily reviewed by OIE.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides assurance that containment leak-tight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits.

"Maintaining containment leakage rates within such limits provides assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of General Design Criteria 52, 53 and 54 of Appendix A to 10 CFR Part 50."

V. REFERENCES

1. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

2. 10 CFR Part 50, Appendix A, General Design Criterion 52, "Capability for Containment Leakage Rate Testing."

3. 10 CFR Part 50, Appendix A, General Design Criterion 53, "Provisions for Containment Testing and Inspection."

4. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Systems Penetrating Containment."

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SECTION 6.3 EMERGENCY CORE COOLING SYSTEM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The RSB reviews the information presented in the applicant's safety analysis report (SAR) regarding the emergency core cooling system (ECCS). The major elements of the review are:

1. Design Bases

The design bases for the ECCS are reviewed to assure that they satisfy applicable regulations, including the general design criteria and the amendments to 10 CFR 50 regarding ECCS acceptance criteria issued by the Commission on December 28, 1973 (Ref. 1).

2. Design

The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

3. Test Program

The preoperational and initial startup test programs for the ECCS are reviewed to determine if they are sufficient to confirm the performance capability of the ECCS. The need for special design features to permit the performance of adequate test programs is also evaluated as a part of this review.

4. Technical Specifications

The proposed technical specifications are reviewed to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The ECCS is also reviewed to assure that it has the proper seismic and quality group classifications. This aspect of the review is performed as a part of the effort described in

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

Standard Review Plans (SRP) 3.2.1 and 3.2.2. The ECCS system is to be enclosed in structures having the proper seismic classification. This review is done as described in SRP 3.2.2. The structures are reviewed by SEB.

APCSB reviews, as described in SRP 9.2.1, 9.2.2, 9.2.5 and 9.2.6, those auxiliary systems essential for ECCS operation (service water system, component cooling system, ultimate heat sink, and condensate storage facility) and assesses the capability of these systems to perform all functions required by the ECCS. The APCSB will supply, on request, evaluations of portions of the power conversion systems (e.g., steam supply lines, steam generators, feedwater systems) which interface with the reactor coolant system in such as way as to influence the course of a loss-of-coolant accident (LOCA) for a particular plant.

The EICSB, as described in SRP 7.3 and 8.3.1, reviews the adequacy of ECCS-associated controls and instrumentation with regard to features of automatic actuation, remote sensing and indication, remote control, and emergency onsite power.

The MEB, as described in SRP 3.9.3, reviews the loading combinations (operational, LOCA, and seismic) and the associated stress limits. On a generic basis, the MTEB reviews the thermal shock effect of water injected into the primary coolant system from the ECCS. The CSB, as described in SRP 6.2.1.3, reviews the analyses of mass and energy released to the containment following a LOCA to determine acceptability of the containment backpressure used in the ECCS capability studies.

The MEB and APCSB review the effects of pipe breaks both inside and outside containment on ECCS. This review includes the effects of pipe whip, jet impingement forces, and any environmental conditions created.

The ability of the ECCS to mitigate the consequences of a spectrum of loss-of-coolant accidents is reviewed by RSB under SRP 15.6.5.

II. ACCEPTANCE CRITERIA

The general objective of the review of the ECCS is to determine that the system meets the applicable general design criteria (GDC), the ECCS acceptance criteria of 10 CFR §50.46, and the intent of applicable regulatory guides.

In regard to the ECCS acceptance criteria (Ref. 1), the five major performance criteria deal with:

1. Peak cladding temperature.
2. Maximum calculated cladding oxidation.
3. Maximum hydrogen generation.
4. Coolable core geometry.
5. Long-term cooling.

These areas are reviewed as a part of the effort associated with the LOCA analysis (SRP 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS is evaluated under this plan.

The ECCS must meet the requirements of GDC 35 (Ref. 8). The system must have alternate sources of electric power, as required by GDC 17 (Ref. 5), and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident. Further, the ECCS should be designed to perform its function considering simultaneous LOCA and seismic loadings.

The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of GDC 36 (Ref. 9). The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37 (Ref. 10).

The portions of the protection system associated with ECCS must meet the requirements of GDC 20 (Ref. 6) and GDC 27 (Ref. 7) and should conform to the recommendations of Regulatory Guide 1.47 (Ref. 17). The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.

The design of the ECCS must be in accord with GDC 2 (Ref. 2); GDC 4 (Ref. 3); and GDC 54 (Ref. 11); and should conform to the recommendations of Regulatory Guide 1.1 (Ref. 13), Regulatory Guide 1.29 (Ref. 15); Regulatory Guide 1.46 (Ref. 16); and staff positions on protection against piping failures outside containment (Ref. 21).

All ECCS and instrument lines that penetrate the primary reactor containment must be provided with suitable isolation valves in accordance with the requirements of GDC 56 (Ref. 12) and should meet the recommendations of Regulatory Guide 1.11 (Ref. 14).

Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to GDC 5 (Ref. 4).

The preoperational and initial startup test programs should meet the intent of Regulatory Guide 1.68 (Ref. 19) and Regulatory Guide 1.79 (Ref. 20).

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For operating license (OL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes: (1) the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing; and (2) the preoperational and initial startup test programs, to assure that they meet the intent of Regulatory Guides 1.68 and 1.79 (Refs.19 and 20).

Much of the review described below is generic in nature and is not performed for each plant. That is, the RSB reviewer compares the ECCS design and parameters to those of previously reviewed plants and then devotes the major portion of the review effort to those areas where the application is not identical to previously reviewed plants. The following steps are taken by the RSB reviewer to determine that the acceptance criteria of Section II have been met. These steps should be adapted to CP or OL reviews as appropriate.

1. The relationship of the system under review to other previously approved plants is established. Systems or design features claimed to be identical or equivalent to those of previously approved plants are confirmed to be identical or equivalent.

2. Piping diagrams are reviewed to evaluate the functional reliability of the system in the event of single failures. That is, by referring to piping and instrumentation diagrams, the existence of the redundancy required by the criteria is confirmed.

3. The significant design parameters (e.g., pump net positive suction head, pump head vs. flow, accumulator volume and pressure, water storage volume, system flow rate and pressure, etc.) are examined for each component to confirm that these parameters satisfy operating requirements and the recommendations of Regulatory Guide 1.1 (Ref. 13).

4. The piping and instrumentation diagrams are checked to see that essential ECCS components are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III).

5. The ECCS design is reviewed to confirm that the system can function in post-accident environments, considering possible mechanical effects, missiles, and the pressure, temperature, moisture, radioactivity, and chemical conditions resulting from LOCA. Protection against valve motor flooding should be confirmed by the RSB reviewer. In regard to possible mechanical effects and missiles, the RSB reviewer confirms that appropriate reviews have been made as described in other review plans, and that the ECCS has been found, in these reviews, to meet the requirements of GDC 2 and 4, the recommendations of Regulatory Guides 1.29 and 1.46, and staff positions on pipe breaks outside containment (Refs. 2, 3, 15, 16 and 21). Regarding the effects of pressure, temperature, etc., the RSB reviewer should confirm that, prior to installation, representative active components used in the ECCS will be proof-tested under environmental conditions and for time periods representative of the most severe operating conditions to which they may be subjected.

6. The criteria, supporting analyses, plant design provisions, and operator actions that will be taken are reviewed to ensure that there will not be unacceptably high concentrations of boric acid in the core region (resulting in precipitation of a solid phase) during the long-term cooling phase following a postulated LOCA.

7. The ECCS design is reviewed to confirm that there are provisions for maintenance of the long-term coolant recirculation and decay heat removal systems, e.g., pump or valve overhaul, in the post-LOCA environment (including consideration of radioactivity).

8. The availability of an adequate source of water for the ECCS is confirmed, and the source volume, location, and susceptibility to failure (e.g., freezing) are evaluated. (RSB will request APCSB review as required.) In PWR's, the piping from the water source to the ECCS safety injection pumps are evaluated for conformance with RSB 6-1 (Ref. 22).

9. The ECCS flow paths are reviewed to determine the extent to which flow from the ECCS pumps is diverted as a backup feature to other safeguards equipment (e.g., RHR, containment spray). The reviewer should confirm that the remaining portion of the flow provides abundant core cooling, despite the most severe single failure that affects ECCS flow.

10. For a boiling water reactor (BWR), the reactor coolant automatic depressurization system is reviewed to confirm the capability to satisfy LOCA pressure relief functions, including consideration of a single failure.

11. The design of ECCS injection lines is reviewed to confirm that the isolation provisions at the interface with the reactor coolant system are adequate. The number and type of valves used to form the interface between low pressure portions of the ECCS and the reactor coolant system must provide adequate assurance that the ECCS will not be subjected to a pressure greater than its design pressure. This may be accomplished by any of the following provisions:

    a. One or more check valves in series with a normally closed motor-operated valve. The motor-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

    b. Three check valves in series.

    c. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and the testing is performed at least annually.

12. ECCS piping and instrument lines that penetrate containment are reviewed to confirm that there are appropriate containment isolation measures, in accordance with Regulatory Guide 1.11 (Ref. 14).

13. Motor-operated isolation valves in ECCS lines connecting the accumulators to the reactor coolant system in a pressurized water reactor (PWR) are reviewed to ensure that adequate provisions are made against inadvertent isolation.

14. The capacity and settings of relief valves provided for the ECCS to satisfy system overpressure protection requirements are reviewed. In particular, for PWR's, the reviewer confirms that the accumulator relief valves have adequate capacity so that leakage from the reactor coolant system will not jeopardize the integrity of the accumulators.

15. The reviewer confirms that the design has provisions to assure that ECCS injection lines are maintained in a filled condition, to preclude the possibility of a water hammer when injection flow is initiated.

16. The reviewer confirms that no component or feature of the ECCS in one reactor facility on a multiple plant site is shared with the ECCS in another facility, or that shared features clearly meet the requirements of GDC 5 (Ref. 4).

17. The reviewer confirms that within an individual reactor facility, any components shared between the ECCS and other systems (e.g., coolant makeup systems, residual heat removal systems, containment cooling systems) satisfy engineered safeguard feature design requirements and that the ECCS function of the shared component is not diminished by the sharing.

18. The reviewer confirms that ECCS components located exterior to the reactor containment are housed in a structure which, in the event of leakage from the ECCS, permits venting of releases through iodine filters designed in accordance with Regulatory Guide 1.52 (Ref. 18).

19. The complete sequence of ECCS operation from accident occurrence through long-term core cooling is examined to see that a minimum of manual action is required, and where manual action is used, a sufficient time (greater than 20 minutes) is available for the operator to respond.

20. The reviewer confirms that long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. If an intermediate heat transport system, such as the component cooling water system, is used to provide long-term cooling capability, the system must be designed and constructed to an appropriate group classification, must be seismic Category I, and must be capable of sustaining a single active or passive failure without loss of function. Intermediate systems that conform to the staff positions on piping failures outside containment (Ref. 21) are considered to meet these requirements. (RSB will request APCSB review as required.)

21. The RSB reviewer consults with the EICSB reviewer to:

a. Confirm that the power requirements of the ECCS, including the timing of electrical loads, are compatible with the design of onsite emergency power systems, both a-c and d-c.

b. Confirm that there are sufficient instrumentation and controls available to the reactor operator to provide adequate information in the control room to assist in assessing post-LOCA conditions, including the more significant parameters such as coolant flow, coolant temperature, and containment pressure. If ECCS flow is diverted as a backup to other safeguards systems, the reviewer confirms that instrumentation and controls are available to provide sufficient information in the control room to determine that adequate core cooling is being provided.

c. Confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required, that suitable interlocks are provided, which do not impair separation of power trains or inhibit the required valve motions, and that instrumentation and controls have sufficient redundancy to satisfy the single failure criterion.

22. Analyses are provided by the applicant in Chapter 15 of the SAR to assess the capability of the ECCS to meet functional requirements. These analyses are reviewed by the RSB, as described in SRP 15.6.5, to determine conformance to the acceptance criteria for ECCS. However, the following portions of the review of ECCS response in loss-of-coolant accidents are performed by the RSB reviewer under this plan:

a. The lower limit of break size for which ECCS operation is required is established; i.e., the maximum break size for which normal reactor coolant makeup systems can maintain reactor pressure and coolant level is determined. The capability of the ECCS to actuate and perform at this lower limit of break size is confirmed.

b. The reviewer confirms that the analyses take into account a variety of potential locations for postulated pipe breaks, including ECCS injection lines.

c. The reviewer confirms that the analyses take into account a variety of single active failures. The reviewer should keep in mind that different single failures may be limiting, depending on the particular break location and break size postulated.

d. The ECCS component response times (e.g., for valves, pumps, power supply) are reviewed to confirm that they are within the delay times used in the accident analyses.

e. The ECCS design adequacy for all modes of reactor operation (e.g., full power, low power, hot standby, cold shutdown, partial loop isolation) is confirmed.

23. The proposed plant technical specifications are reviewed to:

a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when ECCS equipment is inoperable due to repairs and maintenance. The means of indicating that safety systems have been bypassed or are inoperable should be in accordance with Regulatory Guide 1.47 (Ref. 17).

b. Confirm that the limiting conditions of operation ensure that the specified operating parameters (minimum poison concentrations, minimum coolant reserve in storage, etc.) are within the bounds of the analyzed conditions.

c. Verify that the frequency and scope of periodic surveillance testing is adequate.

24. The reviewer confirms that the design provides the capability for periodically demonstrating that the system will operate properly when an accident signal is received. That is, it should be demonstrated by an applicant that pumps and valves operate on normal and emergency power and that water pressure and flow are as designed when the plant is operating (periodic system surveillance). When the plant is shut down for refueling, the system should be tested for delivery of coolant to the vessel.

25. The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guides 1.68 and 1.79 (Refs. 19 and 20). At the OL stage, this aspect of the review is to assure that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see par. C.2.b of Regulatory Guide 1.68).

The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the ECCS will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the ECCS design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the ECCS. If the proposed ECCS differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the CP stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

26. Information is provided to other branches in those areas where the RSB has a secondary review responsibility that is not explicitly covered by steps 1-25, above. These additional areas of RSB secondary review responsibility include:

a. Confirmation that the LOCA forcing functions (blowdown loads) used to conduct the system dynamic analysis are representative of the most adverse LOCA loadings. This information is used by MEB as described in SRP 3.9.3.

b. Determining the acceptability of the mass and energy release rates to the containment following a LOCA for use in the minimum containment backpressure analyses performed by CSB, as described in SRP 6.2.1.3. CSB will then inform RSB of the acceptability of the containment backpressure used in the ECCS performance capability studies.

c. Identification (to APCSB) of essential auxiliary systems and components associated with the ECCS that are required for accident conditions, and accident cooling load functional requirements and minimum time intervals.

d. Identification (to APCSB) of process sampling system functional performance requirements for the reactor coolant system during and subsequent to postulated accident conditions.

e. Identification (to APCSB) of essential components associated with the main steam and auxiliary feedwater systems that are required to operate during and following accident conditions.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report. (For completeness, this evaluation finding includes the RSB review effort described in SRP 15.6.5.)

"The emergency core cooling system (ECCS) includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core following a loss-of-coolant accident. The scope of review of the ECCS for the _____ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses have been reviewed and have been found to conform to Commission regulations as set forth in the general design criteria, and to applicable regulatory guides and staff technical positions. The system was found capable of performing its function with only onsite electric power or with only offsite electric power, assuming the most restrictive single active failure.

"The applicant provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR §50.46. The five major performance criteria deal with:

(1) Peak cladding temperature.
(2) Maximum calculated cladding oxidation.
(3) Maximum hydrogen generation.
(4) Coolable core geometry.
(5) Long-term cooling.

"Based on our review of the applicant's analysis, we conclude that the proposed ECCS satisfies the acceptance criteria.

"The staff concludes that the design of the emergency core cooling system conforms to the Commission's regulations and to applicable regulatory guides and staff technical positions, and is acceptable."

V. REFERENCES

1. 10 CFR §50.46,"Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," issued by the Commission December 28, 1973; Federal Register, Vol. 39, No. 3, January 4, 1974.

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

5. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."

6. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

7. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control System Capability."

8. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

9. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System."

10. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

11. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

12. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

13. Regulatory Guide 1.1, "Net Position Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

14. Regulatory Guide 1.11, "Instrument Lines Penetrating Containment."

15. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

16. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

17. Regulatory Guide 1.47, "Bypass and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

18. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

19. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

20. Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."

21. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid Systems Piping Outside Containment," attached to SRP 3.6.2.

22. Branch Technical Position RSB 6-1, "Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps," attached to SRP 6.3.

BRANCH TECHNICAL POSITION RSB 6-1

PIPING FROM THE RWST (OR BWST) AND CONTAINMENT SUMP(S) TO THE SAFETY INJECTION PUMPS

A. Background

Current PWR's utilize the refueling water storage tank (RWST) or the borated water storage tank (BWST) as the sole source of water for the safety injection pumps during the first twenty to forty minutes of any accident that trips a safety injection signal. Since acceptable results of safety analyses of the accidents are based on the operation of a minimum number of these pumps, interruption of this water supply for even a short period of time could result in unacceptably high fuel and cladding temperatures if the safety injection pumps fail because of cavitation or over heating.

General Design Criteria 35 requires that the emergency core cooling system have suitable redundancy in components and features and suitable interconnections to assure the system safety function can be accomplished assuming a single failure. The principal problem appears to be a definition of single failure. A recent draft of ANSI N658, "Single Failure Criteria for PWR Fluid Systems" defines an active failure as:

(a) "An active failure is a malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function upon demand."

(b) "Spurious action of a powered component originating within its actuation system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious action."

This branch position on the availability of the RWST is based on the above criteria and the recognition that water supplied from the RWST to the ECCS system is absolutely essential in the event of a LOCA.

B. Branch Position

1. The single active failure criterion defined in (a) and (b) above will be applied in evaluating the design of the piping systems that connect the safety injection pumps to the RWST (BWST) and the containment sumps.

2. The piping systems, including valves, shall be designed to satisfy the requirements listed below without the need to disconnect the power to any valve.

3. The valves and piping between the RWST (or BWST) and the safety injection pumps must be arranged so that no single failure will prevent the minimum flow to the core required to satisfy 10 CFR 50.46.

4. The valves and piping between the RWST (or BWST) and safety injection pumps must be arranged so that no single active failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied.

5. The valves and piping that connect the RWST (or BWST) and the containment sump(s) to the safety injection pumps must be arranged so as not to preclude automatic switchover from the injection mode of ECCS operation to recirculation cooling from the sump. These piping systems must be arranged so that the differential pressure between the sump and the RWST (or BWST), even if there is a single active failure, will not result in a loss of core cooling or a path that permits release of radioactive material from the containment to the environment.

C. Implementation

1. CP's Under Review and Future CP Reviews

The proposed position will be applied to all CP reviews for which an SER was not published prior to April 16, 1975. It is expected that all of the elements of the proposed position will be applied for such reviews. Taking this position on CP's would eliminate the need for various schemes such as locking out power to valves located in the line between the various ECCS pumps and refueling water storage tank.

2. OL's Under Review

For operating licenses that are presently under review and OL's to be reviewed in the future that are not covered by item 1, the proposed position will not be completely applied. Specifically, locking out power to valves will be permitted. For most plants it is expected that this will be sufficient to meet the single failure criteria. However, in other plants changes to the piping and valving arrangements may be required to satisfy the single failure criteria.

3. Plants Under Construction

These plants will be handled as discussed in item C2. It is expected, however, that we will discuss the proposed position with each of the applicable PWR vendors. It will be obvious to the vendors which plants now under construction may have a problem. Then a generic review may be conducted for those plants that have a severe problem.

4. Operating Plants

All of the operating plants are being evaluated as an ongoing part of the current ECC review. The review should be conducted as discussed in item C2 to assure that these plants meet the essential parts of the proposed position.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.4 HABITABILITY SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Hydrology-Meteorology Branch (HMB)
Auxiliary Systems Branch (ASB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The control room ventilation system and control building layout and structures, as described in the applicant's safety analysis report (SAR), are reviewed with the objective of assuring that plant operators are adequatel, tected against the effects of accidental releases of toxic or radioactive gases. A further objective is to assure that the control room can be maintained as the center from which emergency teams can safely operate in the case of a design basis radiological release. To assure that these objectives are accomplished the following items are reviewed:

1. The zone serviced by the control room emergency ventilation system is examined to ascertain that all critical areas requiring access in the event of an accident are included within the zone (control room, kitchen, sanitary facilities, etc.) and to assure that those areas not requiring access are generally excluded from the zone.

2. The capacity of the control room in terms of the number of people it can accommodate for an extended period of time is reviewed to confirm the adequacy of emergency food and medical supplies and self-contained breathing apparatus and to determine the length of time the control room can be isolated before CO_2 levels become excessive.

3. The control room ventilation system layout and functional design is reviewed to determine flow rates and filter efficiencies for input into the AAB analyses of the buildup of radioactive or toxic gases inside the control room, assuming a design basis release. Basic deficiencies that might impair the effectiveness of the system are examined. In addition, the system operation and procedures are reviewed. The ASB has primary responsibility in the system review area under Standard Review Plan (SRP) 9.4.1. The ASB is consulted when reviewing hardware and operating procedures.

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Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

4. The flow rates and iodine removal efficiencies used in the analysis are obtained from the ETSB (see SRP 6.5.1).

5. The physical location of the control room with respect to potential release points of hazardous airborne materials (SAR Chapter 2 and other pertinent chapters) is reviewed to determine the location and source strength of radioactive, toxic, or noxious materials. The layout of the control building is reviewed to assure that airborne materials will not enter the control room from corridors or ventilation ducts, etc. Estimates of dispersion of airborne contamination are made in conjunction with HMB.

6. Radiation shielding provided by structural concrete is analyzed to determine the effectiveness of shielding and structure surrounding the control room. The control building layouts are checked to see if radiation streaming through doors (or other apertures) or from equipment might be a problem.

7. Independent analyses are performed to determine whether dose values or toxic gas concentrations remain below recommended levels. The HMB provides meteorological input and checks the X/Q values for the control room location.

II. ACCEPTANCE CRITERIA

1. Control Room Emergency Zone
   See Section III.1 of this plan.

2. Control Room Personnel Capacity
   Food, water, and medical supplies should be sufficient to maintain the emergency team (at least 5 men) for 5 days.

3. Ventilation System Criteria (See III.3 of this plan)
   The following criteria deal with the verification of acceptable system performance and assurance of system availability:

   a. Isolation Dampers - Dampers used to isolate the control zone from adjacent zones or the outside must be leaktight. This may be accomplished by using low leakage dampers or valves. The degree of leaktightness should be documented in the SAR.

   b. Single Failure - A single failure of an active component should not result in loss of the system's functional performance. All the components of the control room emergency filter train will be considered active components. See Appendix A for criteria regarding valve or damper repair.

   c. Pressurization Systems - Systems that will pressurize the control room during a radiation emergency should meet the following requirements:

      (1) Those systems having pressurization rates of greater than or equal to 0.5 volume changes per hour will require periodic (every 18 months) verification that the makeup is $\pm$ 10% of design value. During plant construction or

after any modifications to the control room that might significantly affect its capability to maintain a positive pressure, measurements should be taken to verify that the control room is pressurized to at least 1/8-inch water gauge relative to all surrounding air spaces (while applying the design makeup air rate).

(2) Those systems having pressurization rates of less than 0.5 and equal to or greater than 0.25 volume changes per hour will have identical testing requirements as indicated in (1), above. In addition, at the CP stage an analysis must be provided (based on the planned leaktight design features) that ensures the feasibility of maintaining 1/8-inch water gauge differential with the design makeup air flow rate.

(3) Those systems having pressurization rates of less than 0.25 volume changes should meet all the requirements for (2), above, except that periodic verification of control room pressurization (every 18 months) will be required.

4. Toxic Gas Protection

Self-contained breathing apparatus for the emergency team (at least 5 men) should be on hand. A six-hour onsite bottled air supply should be available with unlimited offsite replenishment capability from nearby location(s). Refer to References 3 through 6, and see Section III.3 of this plan.

5. Emergency Standby Filters

See Standard Review Plan 6.5.1 for acceptance criteria for control room ESF systems. Credit for iodine removal efficiencies will be given in accordance with Regulatory Guide 1.52. Filter efficiencies for systems not covered by Regulatory Guide 1.52 will be determined on a case-by-case basis by ETSB.

6. Relative Location of Source and Control Room

In general, the control room inlets must be so placed in relation to the location of potential release points as to minimize control room contamination in the event of a release. Specific criteria as to radiation and toxic gas sources are as follows:

a. Radiation Sources

As a general rule the control room ventilation inlet should be separated from the major potential release points by at least 100 feet laterally and by 50 feet vertically. However, the actual minimum distances must be based on the dose analyses. Refer to Section III of this plan and Reference 7 for further information.

b. Toxic gases

The minimum separation distance is dependent upon the gas in question, the container size, and the available control room protection provisions. Refer to Regulatory Guide 1.78 (Ref. 3) for general guidance and to Regulatory Guide 1.95 (Ref. 4) for specific acceptable design provisions related to chlorine.

7. Radiation Shielding

See discussion of General Design Criterion 19 below.

8. Radioactive and Toxic Gas Hazards

a. Radiation Hazards

The dose guidelines (see General Design Criterion 19, Appendix A of 10 CFR Part 50) used in approving emergency zone radiation protection provisions are as follows:

(1) Whole body gamma: 5 rem

(2) Thyroid: 30 rem

(3) Beta skin dose: 30 rem*

The whole body gamma dose consists of contributions from airborne radioactivity inside and outside the control room, as well as direct shine from fission products inside the reactor containment building.

b. Toxic Gases

For acceptance purposes, three exposure categories are defined: protective action exposure (2 minutes or less), short-term exposure (between 2 minutes and 1 hour), and long-term exposure (1 hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance: there should be no chronic effects from exposure, and acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of medication other than the use of self-contained breathing apparatus.

The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows:

(1) Long-term limit (1 hour or greater): use a limit assigned for occupational exposure (40-hour week).

(2) Short-term limit (2 minutes to 1 hour): use a limit that will assure that the operator will not suffer incapacitating effects after a 1-hour exposure.

*Credit for the beta radiation shielding afforded by special protective clothing and eye protection is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. The skin and thyroid dose levels are to be used only for judging the acceptability of the design provisions for protecting control room operators under postulated design basis accident conditions. They are not to be interpreted as acceptable emergency doses. The dose levels quoted here are derived for use in the controlled plant environment and should not be confused with the conservative dose computation assumptions used in evaluating exposures to the general public for the purposes of comparison with the guideline values of 10 CFR Part 100.

(3) Protective action limit (2 min. or less) use a limit that will assure that the operator will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes.

The protective action limit is used to determine the acceptability of emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. (They are also used in those cases where the toxic levels are such that emergency zone isolation without use of protective gear is sufficient.) As an example of appropriate limits, the following are the three levels for chlorine gas:

Long-term:	1 ppm by volume
Short-term:	4
Protective action:	15

(See Reference 3 for protective action levels for other toxic gases.)

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed for other plants and whether items of special safety significance are involved.

1. Control Room Emergency Zone

The reviewer checks to see that the zone includes the following:

a. Instrumentation and controls necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file.

b. The computer room, if it is used as an integral part of the emergency response plan.

c. The shift supervisor's office.

d. The operators' wash room and the kitchen.

The emergency zone should be limited to those spaces requiring operator occupancy. Spaces such as battery rooms, cable spreading rooms, or any other spaces not requiring continuous or frequent occupancy after a design basis accident (DBA) generally should

be excluded from the emergency zone. Inclusion of these spaces may increase the probability of smoke or hazardous gases entering the emergency zone. They may also increase the possibility of infiltration into the emergency zone, thus decreasing the effectiveness of the ventilation system in excluding contamination. It is advantageous to have the emergency zone located on one floor, with the areas included in the zone being contiguous.

2. Control Room Personnel Capacity

The reviewer checks to see that emergency food and water are provided. Normally, a five-day supply for five men would be sufficient for land-based plants. A medical kit is also helpful. Specific requirements for these items have not been formulated. The air inside a 100,000 cubic foot control room would support five persons for at least six days. Thus, CO_2 buildup in an isolated emergency zone is not normally considered a limiting problem.

3. Ventilation System Layout and Functional Design

This area is a major portion of the review. The procedures are as follows:

a. The type of system proposed is determined. The following types of protection provisions are currently being employed for boiling water reactor (BWR) or pressurized water reactor (PWR) plants:

(1) Zone isolation, with the incoming air filtered and a positive pressure maintained by the ventilation system fans. This arrangement is often provided for BWRs having high stacks. Air flow rates are between 400 and 4000 cfm.

(2) Zone isolation, with filtered recirculated air. This arrangement is often provided for BWRs and PWRs with roof vents. Recirculation rates range from 2,000 to 30,000 cfm.

(3) Zone isolation, with filtered recirculated air, and with a positive pressure maintained in the zone. This arrangement is essentially the same as that in (2), with the addition of the positive pressure provision.

(4) Dual air inlets for the emergency zone. In this arrangement, two widely spaced inlets are located outboard (on opposite sides) of potential toxic and radioactive gas sources. The arrangement guarantees at least one inlet being free of contamination (except under extreme no-wind conditions). It can be used in all types of plants. Makeup air supplied from the contamination-free inlet provides a positive pressure in the emergency zone and thus minimizes infiltration.

(5) Bottled air supply for a limited time. In this arrangement, a flow rate of 400 to 600 cfm is provided from compressed air containers for about one hour, to prevent inleakage. It is used in systems having containments whose internal atmospheric pressure becomes negative within an hour after a DBA (subatmospheric containments).

b. The input parameters to the radiological dose model are determined (see Item 5). The parameters are emergency zone volume, filter efficiency, filtered makeup air flow rate, unfiltered inleakage (infiltration), and filtered recirculated air flow rate.

c. The ventilation system components and the system layout diagrams are examined. The responsible reviewer in the ASB should be consulted if there are questions pertaining to the system design. He will determine if the system meets the single failure criterion as well as other safety requirements (see Standard Review Plan Section 9.4.1). Damper failure and fan failure are especially important. The review should confirm that the failure of isolation dampers on the upstream side of fans will not result in too much unfiltered air entering the control room. The AAB dose analysis results are used to determine how much unfiltered air can be tolerated.

d. The following information may be used in evaluating the specific system types (see Reference 7 for further discussion):

(1) Zone isolation, with filtered incoming air and positive pressure. These systems may not be sufficiently effective in protecting against iodine. The staff allows an iodine protection factor (IPF), which is defined as the time-integrated concentration of iodine outside over the time-integrated concentration within the emergency zone, of 20 to 100 for filters built, maintained, and operated according to Regulatory Guide 1.52 (an IPF of 100 requires deep bed filters). Such systems are likely to provide a sufficient reduction in iodine concentration only if the source is at some distance from the inlets. Thus, in most cases only plants with high stacks (~ 100 m) would meet Criterion 19 with this system. Normally the staff suggests that these systems be modified to allow isolation and operation with recirculated air since only minor ducting changes are necessary.

(2) Zone isolation, with filtered recirculated air. These systems have a greater potential for controlling iodine than those having once-through filters. IPFs ranging from 20 to over 150 can be achieved. These are the usual designs for plants having vents located at containment roof level. A system having a recirculation rate of 5000 cfm and a filter efficiency of 95% would be rated as follows:

Infiltration (cfm)	IPF*
200	25
100	49
50	96
25	191

*Within the range of interest, the iodine protection factor is directly proportional to recirculation flow rate times efficiency.

Infiltration should be determined conservatively. The calculated or measured gross leakage is used to determine the infiltration rate that will be applied in the evaluation of the radiological consequences of postulated accidents. This rate is determined as follows:

(i) The leakage from the control room when pressurized to 1/8-inch water gauge is calculated on the basis of the gross leakage data. One-half of this value is used to represent the base infiltration rate. Component leak rates may be used to calculate gross leakage (see, for example, References 8 and 9).

(ii) The base infiltration rate is augmented by adding to it the estimated contribution of opening and closing of doors associated with such activities as the required emergency procedures external to the control room. Normally, 10 cfm is used for this additional contribution.

(iii) An additional factor that is used to modify the base infiltration rate is the enhancement of the infiltration occurring at the dampers or valves upstream of recirculation fans. When closed, these dampers typically are exposed to a several-inch water gauge pressure differential. This is accounted for by an additional infiltration contribution over the base infiltration at 1/8-inch water gauge.

The use of an infiltration rate that is based on calculation is acceptable except in the case where the applicant has assumed exceptionally low rates of infiltration. In these cases, more substantial verification or proof may be required. For instance, if an applicant submits an analysis that shows a gross leakage rate of less than 0.06 volume changes per hour, the reviewer would require that the gross leakage be verified by periodic tests as described in Regulatory Position C.5 of Regulatory Guide 1.95.

(3) Zone isolation, with filtered recirculated air, and with a positive pressure. This system is essentially the same as the preceding one. However, an additional operational mode is possible. Makeup air for pressurization is admitted. It is filtered before entering the emergency zone. Pressurization reduces the unfiltered inleakage that is assumed to occur when the emergency

zone is not pressurized. Assuming a filter fan capacity of 5000 cfm and a filter efficiency of 95%, the following protection factors result (flows in cfm):

Makeup Air	Recirculated Air	IPF (Assuming No Infiltration)	IPF (Assuming Infiltration*)
400	4600	238	159
750	4250	128	101
1000	4000	96	80

The makeup flow rate should have adequate margin to assure that the control woom will be maintained at a pressure of at least 1/8-inch water gauge. The applicant should indicate that an acceptance test will be performed to verify adequate pressurization. If the makeup rate is less than 0.5 volume changes per hour, supporting calculations are required to verify adequate air flow. If the makeup rate is less than 0.25 volume changes per hour, periodic verification testing is required in addition to the calculations and the acceptance test.

A question that often arises is whether pressurization" or "isolation and recirculation" of the control room is to be preferred. Which design gives the lowest doses depends upon the assumptions as to unfiltered inleakage. Isolation is generally preferred in that it will limit the entrance of noble gases (not filterable) and, in addition, it is a better approach when the accident involves a short-term "puff release." If infiltration is 25 cfm or less, "isolation" would be best in any event.

A second question related to the first involves the method of operation. The following possibilities have been considered:

(i) Automatic isolation with subsequent manual control of pressurization.

(ii) Automatic isolation with immediate automatic pressurization.

The first is advantageous in the case of external puff releases. Simple isolation would minimize the buildup of the unfilterable noble gases. It would also protect the filters from excessive concentrations in the case of a chlorine release. However, the second method does guarantee that infiltration (unfiltered) is reduced to near zero immediately upon accident detection. This would be beneficial inthe case where the contamination transport path

*Normally 10 cfm infiltration is assumed for conservatism. This flow could be reduced or eliminated if the applicant provides assurance that backflow (primarily as a result of ingress and egress) will not occur. This may mean installing two-door vestibules or equivalent.

to the emergency zone is mainly inside the building. Method (i) should be used in the case of a toxic gas release and either method (i) or (ii) should be used in the case of a radiological release, provided Criterion 19 guidelines can be satisfied. (A substantial time delay should be assumed where manual isolation is assumed, e.g., 20 minutes for the purposes of dose calculations.)

(4) Dual air inlets for the emergency zone. Several plants have utilized this concept. The viability of the dual inlet concept depends upon whether or not the placement of the inlets assures that one inlet will always be free from contamination. The assurance of a contamination-free inlet depends in part upon building wake effects, terrain, and the possibility of wind stagnation or reversal. For example, in a situation where the inlets are located at the extreme edges of the plant structures (e.g., one on the north side and one on the south side), it is possible under certain low probability conditions for both inlets to be contaminated from the same point source. Reference 7 presents the interim position for dealing with the evaluation of X/Q's for dual inlet systems.

With dual inlets placed on plant structures are on opposite sides of potential radiation release points (e.g, containment building), and are capable of functioning with an assumed single active failure in the inlet isolation system, the following considerations may be applied to the evaluation of the control room X/Q's:

(i) Dual inlet designs without manual or automatic selection control - Equation (6) of Reference 7 may be used with respect to the least favorable inlet location to estimate X/Q's. The estimated values can be reduced by a factor of two (2) to account for dilution effects associated with a dual inlet configuration. This is based upon the dilution derived from drawing in equal amounts of clean and contaminated air through two open inlets.

(ii) Dual inlet designs limited to manual selection control - Equation (6) of Reference 7 may be used with respect to the more favorable inlet location to estimate the X/Q's. The estimated values can be reduced by a factor of four (4) to account for dilution effects associated with a dual inlet configuration and the relative probability that the operator will make the proper inlet selection. The reduction factor is contingent upon having redundant radiation detectors within each air inlet. The reduction factor is based on the judgment that trained control room operators, in conjunction with radiation alarm indication, will select and close the contaminated air inlet.

(iii) Dual inlet designs with automatic selection control features - Equation (6) of Reference 7 may be used with respect to the more

favorable inlet location to estimate the X/Q's. The estimated values can be reduced by about a factor of ten (10) to account for the ability to select a "clean" air inlet. The actual factor may be somewhat higher if the inlet configuration begins to approach the remote air inlet concept such that the probability of having one clean air inlet is relatively high. Plant configuration and meteorological conditions should be used as the principal basis for reduction factors greater than ten (10). The reduction factor of ten (10) or more is contingent upon having redundant radiation detectors in each inlet and the provisions of acceptable control logic which would be used in the automatic selection of a clean air inlet.

Because damage to the ducting might seriously affect the system capability to protect the operators, the ducting should be seismic Category I and should be protected against tornado missiles. In addition, the number and placement of dampers must be such as to assure both flow and isolation in each inlet assuming one single active component failure. (See Appendix A for information on the damper repair alternative.) The location of the intakes with respect to the plant security fence should also be reviewed.

(5) Bottled air supply for a limited time. In some plant designs the containment pressure is reduced below atmospheric within one hour after a DBA. This generally assures that after one hour significant radioactive material will not be released from the containment. Such a design makes it feasible to maintain the control room above atmospheric pressure by use of bottled air. Periodic pressurization tests are required to determine that the rated flow (normally about 300 to 600 cfm) is sufficient to pressurize the control room to at least 1/8-inch water gauge. The system is also required to be composed of several separate circuits (one of which is assumed to be inoperative to account for a possible single failure). At least one (nonredundant) once-through filter system for pressurization as a standby for accidents of long duration is also desirable.

Compressed air bottles should be protected from tornado missiles or internally-generated missiles and should be placed so as not to cause damage to vital equipment or interference with operation if they fail.

4. Emergency Standby Filters

Refer to Standard Review Plan 6.5.1.

5. Relative Location of Source and Control Room

This review area involves identification of all potential sources of toxic, radioactive, or otherwise potentially hazardous gases and analysis of their transport to the control room. There are three basic categories: DBA radioactive sources, toxic gases such as

favorable inlet location to estimate the X/Q's. The estimated values can be reduced by about a factor of ten (10) to account for the ability to select a "clean" air inlet. The actual factor may be somewhat higher if the inlet configuration begins to approach the remote air inlet concept such that the probability of having one clean air inlet is relatively high. Plant configuration and meteorological conditions should be used as the principal basis for reduction factors greater than ten (10). The reduction factor of ten (10) or more is contingent upon having redundant radiation detectors in each inlet and the provisions of acceptable control logic which would be used in the automatic selection of a clean air inlet.

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5. Relative Location of Source and Control Room

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chlorine, and gases with the potential for being released inside confined areas adjacent to the control room.

a. DBA Radioactive Sources

The LOCA source terms determined in Appendix A to Standard Review Plan 15.6.5 review are referred to and routinely used to evaluate radiation levels external to the control room. The dispersal from the containment or the standby gas treatment vent is determined with a building wake diffusion model. This model is discussed in Reference 7. Other DBAs are reviewed to determine whether they might constitute a more severe hazard than the LOCA. If this is suspected, an additional analysis is performed for the suspect DBAs. The HMB provides the meteorological input and reviews the AAB calculation of X/Q values.

b. Toxic Gases

The applicant is asked to identify those toxic substances stored (or transported) on or in the vicinity of the site which may pose a threat to the reactor operators by producing toxic gases upon accidental release. The method used to determine whether the quantity or location of the toxic material is such as to require closer study is described in Regulatory Guide 1.78 (Ref. 3). This guide also discusses the methods for analyzing the degree of risk and states, in general terms, the various protective measures that could be instituted if the hazard is found to be too great. In the case of chlorine, specific acceptable protective provisions have been determined; these are described in detail in Reference 4.

In summary, the following provisions or their equivalent are required (pertaining to the emergency zone ventilation system):

(1) Quick-acting toxic gas detectors.

(2) Automatic emergency zone isolation.

(3) Emergency zone leaktightness.

(4) Limited fresh air makeup rates.

(5) Breathing apparatus and associated bottled air supply.

(Note that the best solution for a particular case will depend on the toxic gas in question and on the specific ventilation system design.)

c. Confined Area Releases

The reviewer studies the control building layout in relation to potential sources inside the control building or adjacent connected buildings. The following concerns are checked:

(1) Storage locations of CO_2 or other firefighting materials should be such as to eliminate the possibility of significant quantities of the gases entering the emergency zone. (The ASB has the primary responsibility in this area.)

(2) The ventilation zones adjacent to the emergency zone should be configured and balanced to preclude air flow toward the emergency zone.

(3) All pressurized equipment and piping (e.g., main steam lines and turbines) that could cause significant pressure gradients when failed inside buildings should be isolated from the emergency zone by multiple barriers such as multiple door vestibules or their equivalent.

6. Radiation Shielding

Control room operators as well as other plant personnel are protected from radiation sources associated with a normally operating plant by various combinations of shielding and distance. The adequacy of this type of protection for normal operating conditions is reviewed and evaluated by the RAB. To a large extent the same radiation shielding (and missile barriers) also provides protection from design basis accident radiation sources. This is especially true with respect to the control room walls which usually consist of at least 18 inches of concrete. In most cases, the radiation coming from external design basis accident radiation sources is attenuated to negligible levels. However, the following items should be considered qualitatively in assessing the adequacy of control room radiation shielding:

a. Control room structure boundary. Wall, ceiling, and floor materials and thickness should be reviewed. Eighteen inches to two feet of concrete or its equivalent will be adequate in most cases.

b. Radiation streaming. The control room structure boundary should be reviewed with respect to penetrations (e.g., doors, ducts, stairways, etc.). The potential for radiation streaming from accident sources should be identified, and if deemed necessary, quantitatively evaluated. Support from the RAB may be required for some radiation streaming dose calculations.

c. Radiation shielding from internal sources. If sources internal to the control room complex are identified, radiation shielding against them should be reviewed. Typical sources in this category include contaminated filter trains, or airborne radioactivity in enclosures adjacent to the control room.

Evaluations of radiation shielding effectiveness with respect to the above items should be performed using simplified analytical models for point, line, or volume sources such as those presented in References 10 and 11. If more extended analysis is required, analytical support from the RAB should be requested. The applicant's coverage of the above items should also be reviewed in terms of completeness, method of analysis, and assumptions.

7. Independent Analyses

a. Control Room Doses

Although the applicant is required to calculate doses to the control room operators, independent analyses are made by the AAB because of the diversity of control room habitability system designs and the engineering judgment involved in their evaluation. Using the approach indicated in Reference 7, the source terms and doses due to a DBA are calculated. The source terms determined by the AAB's independent analysis of LPZ doses for a LOCA are used. The methods and assumptions for this calculation are presented in Appendix A to Standard Review Plan 15.6.5. The control room doses are determined by estimating the X/Q from the source points to the emergency zone using meteorological input supplied by the HMB, by determining the credit for the emergency zone's protection features, and by calculating the dose. Figure 6.4-1 shows a form which may be used to summarize the information that is needed for the control room dose calculation. The effective X/Q's are used for calculating the doses. The dose is then compared with the guidelines of General Design Criterion 19. If the guidelines are exceeded, the applicant is asked to improve the system. In the event that other DBAs are expected to result in doses comparable to or higher than the LOCA, additional analyses are performed. The limiting accidents are compared with Criterion 19.

b. Other Analyses

Special case analyses are performed when questions are raised about certain potential sources of toxic or radioactive gases. The methods used in these analyses conform to current DBA methods concerning dispersion and dose calculations. Regulatory Guide 1.78 should be consulted by the site analyst to see if nearby facilities could present a potential hazard that requires detailed analysis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

1. If the plant meets Criterion 19, the following statement or its equivalent is made:

"The applicant proposes to meet General Design Criterion 19 of Appendix A to 10 CFR Part 50 by use of concrete shielding and by installing redundant ____ cfm recirculating charcoal filters in the control room ventilation system. These filters will be automatically activated upon an accident signal, high radiation signal, or high chlorine signal. Independent calculations of the potential radiation doses to control room personnel following a LOCA show the resultant doses to be within the guidelines of Criterion 19."

2. If the design is not adequate, the fact is stated. Alternatives such as an increase in the charcoal filter flow rate may be indicated as is given in the example below:

> "The staff has calculated the potential radiation doses to control room personnel following a LOCA. The resultant whole body doses are within the guidelines of Criterion 19. The thyroid dose resulting from exposure to radioactive iodine exceeds the dose guidelines. The applicant will be required to commit to increasing the filtration system size from 2000 cfm to 4000 cfm. This increased filtration will be sufficient to keep the estimated thyroid doses within the guidelines."

3. If special protection provisions for toxic gases are not required, the following statement or its equivalent is made:

> "The habitability of the control room was evaluated using the procedures described in Regulatory Guide 1.78. As indicated in Section 2.2, no offsite storage or transport of chemicals is close enough to the plant to be considered a hazard. There are no onsite chemicals that can be considered hazardous under Regulatory Guide 1.78. A sodium hypochlorite biocide system will be used, thus eliminating an onsite chlorine hazard. Therefore, special provisions for protection against toxic gases will not be required. Self-contained breathing apparatus is provided for the emergency crew to provide assurance of control room habitability in the event of occurrences such as smoke hazards."

4. If special protection provisions are required, compliance or noncompliance with the guidelines of Regulatory Guides 1.78 and 1.95 should be stated.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

2. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

3. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

4. Draft Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

5. Draft Regulatory Guide 8.X, "Acceptable Programs for Respiratory Protection."

6. "Manual of Respiratory Protection Against Airborne Radioactive Material," WASH-1287, U.S. Atomic Energy Commission (1974).

7. K.G. Murphy and K.M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

8. "Leakage Characteristics of Openings for Reactor Housing Components," NAA-SR-MEMO-5137, Atomics International, Div. of North American Aviation, Inc., June 20, 1960.

9. R.L. Koontz, et al., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Trans. Am. Nucl. Soc., November 1961.

10. R.G. Jaeger, et al., eds., "Engineering Compendium on Radiation Shielding," Vol. 1, "Shielding Fundamentals and Methods," Springer-Verlag (1968).

11. N.M. Schaeffer, "Reactor Shielding for Nuclear Engineers," TID-75951, U.S. Atomic Energy Commission.

SECTION 6.4 APPENDIX A

ACCEPTANCE CRITERIA FOR VALVE OR DAMPER REPAIR ALTERNATIVE

The control room ventilation system must meet the criterion that it work properly even with a single failure of an active component. In certain cases, complex valve or damper configurations are required to meet the single failure criterion. For example, assurance of the isolation and operability of each leg of a dual inlet system at various times after a postulated accident could require a four-valve arrangement in which two pairs of series valves are connected in parallel. The mechnical, power, and control components of such arrangements combine to form a rather complex system. Credit will be allowed for an alternative system that allowed the failed valve to be manually repositioned so that it will not interfere with the operation of the system. For example, in the case of a dual inlet system, if credit for repair is given, then two valves in series in each leg of the dual inlet would be acceptable. Where a valve fails closed but meets the criteria given below, credit would be allowed for the valve to be repositioned and locked in an open position.

The approval of the repair option is contingent upon the intrinsic reliability of the internal components of the valve or damper and also upon the ease and ability to overcome the failure of the external actuating components (electrical relays, motors, hydraulic pistons, etc.). The following criteria or their equivalent will be required:

1. The valve or damper components must be listed as to which are considered internal (non-repairable) and which external (repairable). These must be designed to meet the following criteria.

    a. Internal valve components (components that are difficult to repair manually without opening the ductwork) must be judged to have a very low probability of failure. The component design details will be reviewed and characteristics such as simplicity, ruggedness, and suspectibility to postulated failure mechanisms will be considered in arriving at an engineered judgment of the acceptability of the internal component design with respect to reliability. For example, a butterfly valve welded or keyed onto a pivot shaft would be considered a high reliability internal component. Conversely, multiple-blade dampers, actuated by multi-element linkages or pneumatically-operated components internal to the ducts, would be viewed as being subject to failure.

    b. External valve components (components including motors and power supplies that are to be assumed repairable or removable) must be designed to ensure that the failed valve component can be bypassed easily and safely and that the valve can be manipulated into an acceptable position. The electronic components must be isolated from other equipment to assure that the repair operations do not result in further equipment failure.

2. The location and positioning of the valve or damper must permit easy access from the control room for convenient repair, especially under applicable DBA conditions.

3. Appropriate control room instrumentation should be provided for a clear indication and annunciation of valve or damper malfunction.

4. Periodic manipulation of the valve or damper by control room operators should be required for training purposes and to verify proper manual operability of the valve or damper.

5. The need for manual manipulations of the failed valve or damper should not be recurrent during the course of the accident. Manipulation should not occur more than once during the accident. Adjustment or realignment of other parts of the system should be possible from the control room with the failed valve in a fixed position.

6. The time for repair used in the computation of control room exposures should be taken as the time necessary to repair the valve plus a one-half hour margin. No manual correction will be credited during the first two hours of the accident.

7. Compliance with the above criteria should be documented in the SAR whenever the repair option is used.

FIGURE 6.4-1 Summary Sheet for Control Room Dose Analysis

MEMORANDUM TO: (Site Analyst)
(Meteorologist)

cc: L. Hulman
R.W. Houston (Habitability File)

CONCERNING CONTROL ROOM DOSE ANALYSIS FOR

The following summarizes the X/Q's used in determining the control room operator dose for the subject plant:

VENTILATION SYSTEM DESCRIPTION

SKETCH OF SYSTEM (and inlets/sources if applicable)

SUMMATION OF X/Q ANALYSIS

Source/Receptor Type and Distance

S/D Ratio K Factor

Number of 22 1/2° Wind Direction Sectors that Result in Exposure

Central Wind Sector (sector wind is blowing from)

5% Wind Speed (m/sec) 40% Wind Speed (m/sec)

Projected Area of Wake (m^2) 5% X/Q (sec/m^3)

Time	Wind Speed Factor	Wind Direction Factor	Occupancy Factor	Effective X/Q's
0-8 hr	1	1	1	
8-24 hr			1	
1-4 day			0.6	
4-30 day			0.4	

ACTION REQUESTED

Site Analyst

- For your information only
- Please use the effective X/Q's in TACT run and provide control room doses. In addition, please summarize safety system assumptions and indicate their status (interim or final).

Meteorologist

- These are interim X/Q's. Please review to determine their reasonableness.
- These are final X/Q's. Please determine if they are accurate based on your analysis of site data.

Please Contact _____

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SECTION 6.5.1 ESF ATMOSPHERE CLEANUP SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

At the construction permit (CP) stage of review, ETSB reviews the information in the applicant's safety analysis report (SAR) in the areas listed below. At the operating license (OL) stage, the ETSB review consists of confirming the design accepted at the CP stage and evaluating the adequacy of the applicant's technical specifications in these areas. The specific review areas are as follows:

1. The engineered safety feature (ESF) atmosphere cleanup systems designed for fission product removal in post-accident environments. These generally include primary systems, e.g., recirculation (in-containment), and secondary systems, including standby gas treatment systems and the emergency air cleaning systems for the fuel handling building, control room, and shield building and areas containing engineered safety feature components.

2. The system design, design objectives and design criteria. The ETSB reviews the methods of operation and the factors that influence the filtration capabilities of the system, e.g., system interfaces and potential bypass routes. The components included in each atmospheric cleanup system and the seismic design category of each system are reviewed. Redundancy of the atmosphere cleanup systems, the physical separation of the redundant trains, and the volumetric air flow rate of each train are reviewed.

3. The environmental design criteria, the design pressure and pressure differential, relative humidity, maximum and minimum temperature, radiation source term, and the shielding of essential services such as power and electrical control cables associated with the atmosphere cleanup systems.

4. The component design criteria and qualification testing, qualification requirements of demisters, prefilters, and high-efficiency particulate air (HEPA) filters, design requirements of the filter and adsorber mounting frames, system filter housings, and water drains, the adsorbent used for removal of gaseous iodines (in the preliminary safety analysis report, PSAR), the physical properties of

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

the adsorbent and the design of the adsorber section of the filter trains (in the final safety analysis report, FSAR). Provisions to inhibit offdesign temperatures in the adsorber section and the design criteria of the system fans or blowers, ductwork, and housings are also reviewed.

5. Design provisions incorporated in the equipment and features to facilitate operation and maintenance. The design of doors to the filter housings, the spacing of components, alignment and support of filter elements, the spacing of filter elements in the same bank, design of test probes, and provisions for adequate lighting in the filter housing are also reviewed.

6. The design criteria for inplace testing of the air flow distribution to the HEPA filters, dioctyl phthalate (DOP) testing of the HEPA filter sections, and gaseous halogenated hydrocarbon refrigerant testing of the activated carbon adsorber section.

7. The laboratory testing criteria for the activated carbon adsorbent, qualification and batch tests, provisions for obtaining representative adsorbent samples for laboratory testing in order to estimate the amount of penetration of the system adsorbent throughout its service life (PSAR), and the provisions and conditions for each field and laboratory test (FSAR).

The review of the ESF atmosphere cleanup systems involves secondary review evaluations performed by other branches. The conclusions from their evaluations on request are used by ETSB to complete the overall evaluation of the facility. The evaluations provided by the branches are as follows:

ICSB reviews the associated instrumentation including the power supply and electrical distribution systems under SRP Sections 3.11, 7.3, 7.5, and 8.2; AAB calculates the doses that result as a consequence of postulated accidents under the SRP for Chapters 6, 9, and 15 of the SAR.

II. ACCEPTANCE CRITERIA

The installed ESF atmosphere cleanup system is needed to mitigate the consequences of postulated accidents by removing from the atmosphere radioactive material that may be released in the event of an accident. ETSB will accept ESF atmosphere cleanup systems if the following criteria are met:

1. Atmosphere cleanup systems should be designed so that they can operate after design basis accidents (DBA) and retain radioactive material after the DBA.

2. For the system design, ETSB will use the following guidelines for determining acceptability:

a. Each atmosphere cleanup system should be able to prefilter the air, remove moisture ahead of charcoal adsorbers, and remove particulate matter by HEPA filters before and after the charcoal adsorbers.

b. Redundancy of filter systems should be provided, with the trains physically separated so that damage to one system will not cause damage to the other system.

c. All components should be designated as seismic Category I.

d. Individual systems should be limited to a volumetric air flow rate of 30,000 cfm.

e. Each system should be instrumented to signal, alarm, and record pressure drop and flow rate at the control room.

f. The applicable ESF atmosphere cleanup systems should be automatically activated after a DBA unless (1) the atmosphere cleanup system is operating during the time the DBA occurs, or (2) the activation is the result of another ESF signal (i.e., temperature, pressure).

3. For environmental design, ETSB will use the following guidelines to determine acceptability:

a. Expected conditions for the filter system, including maximum pressure and pressure differential, radiation dose rate received by the components, relative humidity, and maximum and minimum temperature should be based on the conditions in a postulated DBA.

b. The radiation source terms should be consistent with the guidelines in Regulatory Guides 1.3, 1.4, and 1.25.

c. Shielding should be provided for essential services such as power and electrical control cables associated with the atmosphere cleanup system.

4. For component design and qualification testing, ETSB will use the following guidelines to determine acceptability:

a. The demisters should be designed, constructed and tested in accordance with the recommendations of Section 5.4 of ANSI N509-1976 (Ref. 6) and meet the Underwriters' Laboratory (UL) Class 1 requirements (Ref. 8).

b. Moisture removal equipment should be capable of reducing the relative humidity of the incoming atmosphere from 100% to 70%.*

*Air heaters should be designed, constructed and tested in accordance with the recommendations of Section 5.5 of ANSI N509-1976 (Ref. 6).

c. If prefilters are provided, they should be designed, constructed and tested in accordance with the recommendations of Section 5.3 of ANSI N509-1976 (Ref. 6).

d. HEPA filters should be designed, constructed and tested in accordance with Section 5.1 of ANSI N509-1976 (Ref. 6).

e. Filter and adsorber mounting frames should be designed, arranged, and constructed in accordance with the recommendations of Section 5.6.3 of ANSI N509-1976 (Ref. 6).

f. Filter housings, including floors and doors, should be designed and constructed in accordance with the recommendations of Section 5.6 of ANSI N509-1976 (Ref. 6).

g. Water drains should be designed in accordance with the recommendations of Section 4.5.8 of ERDA 76-21 (Ref. 7).

h. The adsorbent to be used for adsorbing gaseous iodine (elemental iodine and organic iodides) should be an adsorbent that has been demonstrated to remove the gaseous iodines from air at the required efficiencies referenced in ANSI N509-1976. If impregnated activated charcoal is the adsorbent, the physical properties of the adsorbent should be in accordance with the guidelines of ANSI N509-1976 (Ref. 6). If an adsorbent other than impregnated activated charcoal is proposed, ETSB will review supporting data and accept adsorbents expected to perform equal to, or better than, impregnated activated charcoal.

i. The adsorption unit should be designed for a maximum loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated charcoal.

j. Provisions should be included to inhibit off-design temperatures in the adsorber section. To dissipate heat generated from iodine decay and charcoal oxidation effects, ETSB will consider cooling mechanisms such as low flow air bleed systems and cooling coils. To extinguish ignited charcoal, ETSB will consider water sprays, carbon dioxide injection systems, and liquid nitrogen cooling systems.

k. The system fan, its mounting, and ductwork connections should be designed, constructed and tested in accordance with the recommendations of Section 5.7 and 5.8 of ANSI N509-1976 (Ref. 6).

l. Ductwork should be designed, constructed and tested in accordance with the recommendations of Section 5.10 of ANSI N509-1976 (Ref. 6).

m. Dampers should be designed, constructed and tested in accordance with the recommendations of Section 5.9 of ANSI N509-1976 (Ref. 6).

5. ETSB will accept ESF filter systems that are designed for accessibility of components and ease of maintenance in accordance with the recommendations of Section 2.5 of ERDA 76-21 (Ref. 7) and Section 4.0 of ANSI N509-1976 (Ref. 6) as follows:

   a. Components should be provided with a minimum of three linear feet from mounting frame to mounting frame between banks of components; components to be replaced should be provided with a minimum of three linear feet plus the maximum length of the component.

   b. Provisions should be made for permanent test probes with external connections in accordance with the recommendations of Section 4.11 of ANSI N509 (Ref. 6).

6. For in-place testing, ETSB will use the following guidelines for determining acceptability:

   a. Provisions should be made for visual inspection of the system and all associated components in accordance with the recommendations of Section 5 of ANSI N510 (Ref. 5).

   b. Provisions should be made for testing the air flow distribution upstream of HEPA filters and charcoal adsorbers, and demonstrating uniformity $\pm$ 20% of averaged flow per unit.

   c. Provisions should be made for DOP testing of the HEPA filter sections in accordance with the recommendations of ANSI N510 (Ref. 5).

   d. Provisions should be made for leak-testing the activated carbon adsorber section with a gaseous halogenated hydrocarbon refrigerant in accordance with the recommendations of ANSI N510 (Ref. 5).

   e. Provisions should be made for in-place testing initially, and routinely thereafter. Frequency and testing requirements will be established in the technical specifications.

7. For laboratory testing of activated carbon adsorbent, ETSB will use the following guidelines for determining acceptability:

   a. Qualification and batch tests on new unused adsorbent should be performed in accordance with the guidelines of ANSI N509-1976 (Ref. 6).

b. Provisions should be made for obtaining representative adsorbent samples in order to estimate the amount of penetration of the system adsorbent throughout its service life (PSAR).

c. Provisions should be made for laboratory testing initially, and routinely thereafter. Frequency and testing requirements will be established in the technical specifications.

ETSB will accept the following deviations from the above acceptance criteria for the post loss-of-coolant accident (LOCA) hydrogen purge filtration system:

1. If the calculated dose (sum of the long-term doses from the LOCA and the purge dose at the low population zone outer boundary) is less than the guidelines of 10 CFR Part 100, no filtration system is required.

2. If a radioiodine decontamination factor of 10 or less is needed for the calculated dose to be below Part 100, an atmosphere cleanup system that meets the acceptance criteria listed in Item 5 of Acceptance Criteria in SRP Section 11.3 should be provided.

3. If a radioiodine decontamination factor of greater than 10 is needed for the calculated dose to be below Part 100, the atmosphere cleanup system should meet all of the above acceptance criteria, except for Items 2b and 2c.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. In the ETSB review the plant design is reviewed to determine where ESF units are needed.

2. The ETSB review is carried out by making a detailed comparison of atmosphere cleanup system designs with the acceptance criteria of Section II, above. The capability of a system to remove fission products in the atmosphere after a DBA is reviewed, based on a design loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated charcoal adsorbent. Designs consistent with the guidelines of Regulatory Guide 1.52 will be assigned the system efficiencies for removal of elemental iodine and organic iodides given in Table 2 of Regulatory Guide 1.52 and a system efficiency of 99% for removal of particulates resulting from a DBA. The assigned efficiencies are for Accident Analysis Branch use in accident analyses to calculate offsite doses to the whole body and thyroid.

IV. EVALUATION FINDINGS

ETSB verifies that sufficient information has been provided and that the review is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The ESF atmosphere cleanup systems include the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated design basis accident. The scope of our review included an evaluation of these systems with respect to the guidelines of Regulatory Guide 1.52. We have reviewed the applicant's system descriptions and design criteria for the ESF atmosphere cleanup systems. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the ESF atmosphere cleanup systems to applicable regulations, guides and industry standards. Based on our evaluation, we find the proposed ESF atmosphere cleanup systems are acceptable, and the filter efficiencies given in Table 2 of Regulatory Guide 1.52 are appropriate for use in accident analyses."

V. REFERENCES

1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

3. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

4. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

5. ANSI N510, "Testing of Nuclear Air Cleaning Systems," American National Standards Institute (1975).

6. ANSI N509, "Nuclear Power Plant Air Cleaning Units and Components," American National Standards Institute (1976).

7. ERDA 76-21, "Nuclear Air Cleaning Handbook," Oak Ridge National Laboratory, C. A. Burchsted, I. E. Kahn and A. B. Fuller, March 31, 1976.

8. "Building Materials List," Underwriters' Laboratories, Inc.

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SECTION 6.5.2 CONTAINMENT SPRAY AS A FISSION PRODUCT CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The AAB reviews the containment spray and spray additive system to determine the fission product removal effectiveness of the system whenever the applicant claims a containment air cleanup function for the system. The heat removal and hydrogen mixing functions (where applicable) are reviewed by the CSB. The sump design is also reviewed by the CSB. Long-term pH requirements are reviewed by the AAB under Standard Review Plan (SRP) 6.1.3.

The following areas of the applicant's safety analysis report (SAR) relating to the fission product removal and control function of the containment spray system are reviewed by the AAB:

1. Fission Product Removal Requirement for Containment Spray

Sections of the SAR related to accident analyses, dose calculations, and fission product removal and control are briefly reviewed to establish whether fission product scrubbing of the containment atmosphere for the mitigation of offsite doses following a postulated accident is claimed by the applicant. This review usually covers Sections 6.2.3.1, 6.5.2.1, and 15.X of the SAR (Ref. 1.).

2. Design Bases

The design bases of such containment spray systems are reviewed to determine whether they reflect the requirements placed upon this system by the assumptions made in the accident evaluations of Chapter 15.

3. System Design

The descriptive information concerning the design of the spray system, including any subsystems and supporting systems, is reviewed to familiarize the reviewer with the design and operation of the system. The review includes:

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a. The descriptive information contained in SAR Sections 6.5.2.2, 6.5.2.4, 6.5.2.5, and 6.5.2.6 (to establish the basic design concept), the systems, subsystems, and support systems required to carry out the iodine scrubbing function of the system, and the components and instrumentation employed in these systems.

b. The process and instrumentation diagrams of SAR Section 6.5.2 or 6.2.2 (whichever contains the relevant information).

c. Layout drawings, (plans, elevations, isometrics) of the spray distribution headers, from SAR Section 6.5.2 or 6.2.2.

d. Plan views and elevations of the containment layout.

e. Process and instrumentation diagrams of any ventilation systems operational in the post-accident environment.

4. Testing and Inspections

Section 6.5.2.5 of the SAR is reviewed to establish the details of the pre-operational test to be performed for system verification and the post-operational tests and inspections to be performed for verification of the continued status of readiness of the spray system.

5. Technical Specifications

At the operating license stage, Sections 3.0 and 4.0 of Chapter 16 of the applicant's final safety analysis report are reviewed to establish permissible outage times and surveillance requirements.

II. ACCEPTANCE CRITERIA

1. General Design Criteria

Criterion 41 ("Containment Atmosphere Cleanup"), Criterion 42 ("Inspection of Containment Atmosphere Cleanup Systems"), and Criterion 43 ("Testing of Containment Atmosphere Cleanup Systems") of Appendix A of 10 CFR Part 50 are used as acceptance criteria.

2. Specific Design Requirements for Iodine Removal Function

a. System Operation

The containment spray system should be designed to be initiated automatically by an appropriate accident signal and should be capable of continuous operation thereafter until the design objectives of the system have been achieved. In all cases the operating period should not be less than two hours. In addition, the system should be capable of operation in the recirculation mode on demand, for a period of at least one month following the postulated accident.

b. Coverage of Containment Volume

In order to assure full coverage of the containment volume, the following should be observed:

(1) The spray nozzles should be located as high in the containment as practicable, to maximize the spray drop fall distance.

(2) The layout of the spray nozzles and distribution headers should be such that the cross-sectional area of the containment covered by the spray is maximized and that a nearly homogeneous distribution of spray in the containment volume is produced. At the operating deck level, at least 95% of the cross section of the containment should be covered. Unsprayed regions in the upper containment and, in particular, an unsprayed annulus adjacent to the containment liner should be avoided wherever possible.

(3) In designing the layout of the spray nozzle positions and orientations, the effect of the post-accident atmosphere should be considered, including the effects of post-accident conditions that result in the maximum possible atmosphere density.

c. Promotion of Containment Mixing

Because the effectiveness of the containment spray system depends on a well-mixed containment atmosphere, all design features enhancing post-accident mixing should be considered. Where necessary, forced air ventilation should be provided to avoid stagnant air regions.

d. Spray Nozzles

The nozzles used in the containment spray system should be of a design that minimizes the possibility of clogging while producing drop sizes effective for iodine absorption. The nozzles should not have internal moving parts such as swirl vanes, turbulence promoters, etc. They should not have orifices or internal restrictions which would narrow the flow passage to less than 1/4 inch in diameter.

e. Injection Spray Solution

The partition of iodine between liquid and gas phases is enhanced by the alkalinity of the solution. The spray system should be design such that the spray solution maintains the highest possible pH, within materials compatibility constraints. This requirement is satisfied by a spray pH in the range of 8.5 to 11.0. Iodine scrubbing credit is given for spray solutions whose chemistry, including any additives, has been demonstrated to be effective for iodine absorption and retention under post-accident conditions. Both theoretical and experimental verification are required. The following solutions have been shown to meet these requirements:

(1) Boric acid solution, 1500 - 2500 ppm boron.

(2) Boric acid solution buffered with NaOH to a pH of 8.5 - 11.0.

(3) Boric acid solution with a minimum of 1.0% by weight of thiosulfate, buffered with NaOH to a pH of 8.5 - 11.0.

(4) Trace level hydrazine (50 ppm nominal) solution.

f. Containment Sump Mixing

The containment sump should be designed to promote mixing of emergency core cooling system (ECCS) and spray solutions. Drains to the engineered safety features (ESF) sump should be provided for all regions of the containment which would collect a significant quantity of the spray solution. Alternatively, allowance should be made for "dead" volumes in the determination of sump pH and the quantities of additives injected.

g. Containment Sump Solutions

The pH of the aqueous solution collected in the containment sump after completion of injection of containment spray and ECCS water, and all additives for reactivity control, fission product removal, or other purpose, should be maintained at a high level. The equilibrium sump pH, after mixing and dilution with the primary coolant and ECCS injection, should be above 8.5 for sodium hydroxide and sodium thiosulfate sprays. A pH value exceeding 8.5 provides assurance that significant evolution of iodine does not occur.

h. Storage of Additives

The design should provide facilities for the long-term storage of all spray additives. These facilities should be designed such that the additives required to achieve the design objectives of the system are stored in a state of continual readiness whenever the reactor is critical during the design life of the plant. The storage facilities should be designed such that freezing, precipitation, chemical reaction, and decomposition of additives are prevented. For NaOH storage tanks, heat tracing of tanks and piping is required whenever exposure to temperatures below 40°F is predicted. An inert cover gas should be provided for solutions with an NaOH concentration of 30% by weight or higher.

i. Single Failure

The system should be able to function effectively and meet all the above criteria with a single failure of an active component in the spray system, any of its subsystems, or any of its support systems. The system is considered functional with respect to iodine removal if it is capable of delivering the design spray flow rate with the additive concentration within the acceptable range as determined above.

3. Testing

Tests should be performed to demonstrate the spray systems, as installed, meet all design requirements for an effective iodine scrubbing function. Such tests should include verification of:

a. Freedom of the containment spray nozzles from obstructions.

b. Capability of the system to deliver the required spray flow.

c. Capability of the system to deliver the required spray additives within the specified range of concentrations.

4. Technical Specifications

The technical specifications should specify appropriate limiting conditions for operation (LCO's), tests, and inspections to provide assurance that the system is capable of its design function whenever the reactor is critical. These specifications should include:

a. The operability requirements for the system, including all active and passive devices, as a limiting condition for operation (with acceptable outage times). The following should be specifically included:

(1) Containment spray pumps.

(2) Additive pumps (if any).

(3) Additive mixing devices (if any).

(4) Additive quantity and concentration in the additive storage tanks.

(5) Nitrogen or other inert gas pressure in the additive storage tanks.

b. Periodic inspection and sampling of the contents of the additive tanks to confirm that the additive quantity and concentrations are within the limits established by the system design.

c. Periodic testing and exercising of the active components of the system and verification that essential piping and passive devices are free of obstructions.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The review of the fission product removal function of the containment spray system follows the procedure outlined below.

The reviewer determines whether the containment spray system is used for fission product removal purposes. (Chapter 15 of the SAR should be reviewed to establish whether a fission product removal function for the containment spray system is assumed in accident dose evaluations.)

If the containment spray system is not used for dose mitigation purposes, i.e., if it is used for heat removal only, no further review is required. (The CSB reviews the heat removal and hydrogen mixing aspects of the containment spray system.) If the containment spray system is designed to reduce the concentrations of fission products in the containment, the capability of the system to function effectively as a fission product removal system is

reviewed. If, as a result of the review, system modifications are required, the CSB is informed of the required modifications for integration with any other requirements placed on the containment spray system.

1. System Design

Review of the system design includes an examination of the components and design features necessary to carry out the iodine scrubbing function, including:

a. Spray Chemistry

The forms of iodine for which spray removal credit is claimed in the accident analyses (SAR Chapter 15) are established. Containment spray systems may be designed for removal of iodine in the vapor (elemental) form, in the form of organic compounds, and in the form of iodine adsorbed on airborne particulate matter.

The systems or subsystems required to carry out the iodine scrubbing function of the containment spray, such as the spray system, recirculation system, spray additive system, and water source are identified.

The conceptual designs of the systems involved are reviewed:

(1) To determine the chemical additive and to ascertain the effectiveness of the additive for elemental and organic iodine removal by comparison with additives of proven effectiveness (see acceptance criteria in Section II) or by review of theoretical and experimental verifications supplied for new additives.

(2) To ascertain that the range of additive concentrations is within the limits listed in the acceptance criteria of Section II or that adequate justification is supplied for the iodine removal and retention effectiveness, radiolytic and pyrolytic decomposition, corrosion, and solidification and precipitation behavior of the chemical additives for the range of concentrations encountered. The concentrations in the storage facility, the chemical addition lines, the spray solution injection, the containment sump solution, and the recirculation spray solution should be examined. The extremes of the additive concentrations should be determined with the most adverse combination of ECCS, spray, and additive pumps (if any) assumed to be operating, and a single active failure of pumps or valves should be considered.

b. System Operation

The time and method of system initiation, including additive addition, is reviewed to confirm that the acceptance criteria of Section II are met. Automatic initiation of spray and spray additive flow, without mechanical delays or manual overrides, is required. The system operation should be continuous until the iodine removal objectives of the system are met. If a switch-over from the injection to a recirculation mode of operation is required, the reviewer should confirm that all

requirements listed in the acceptance criteria, particularly those concerning spray coverage and solution pH, are met during the recirculation phase.

c. Spray Distribution and Containment Mixing

The number and layout of the spray headers used to distribute the spray flow in the containment are reviewed. The reviewer verifies that the layout of the headers assures coverage of essentially the entire cross-section of the containment with spray, under minimum spray flow conditions. The effect of the high temperature and pressure conditions in the containment on the spray droplet trajectories should be taken into account in determining the area covered by the spray.

The layout of the containment and forced ventilation systems (safety-grade) operating after the loss-of-coolant accident (LOCA) are reviewed to determine any areas of the containment free volume that are not sprayed. The rate of mixing (either through forced ventilation or by demonstrated convective mixing) between the sprayed and unsprayed compartments of the containment is evaluated. The containment may be considered a single, well-mixed volume provided the spray covers regions comprising at least 90% of the containment volume and provided a ventilation system is available for adequate mixing of any unsprayed compartments.

d. Spray Nozzles

The design of the spray nozzles is reviewed to confirm that the spray nozzles are not subject to clogging from debris entering the recirculation system through the sump screens.

e. Sump Mixing

The mixing of the spray water containing the chemical additive and water without additive (such as spilling ECCS coolant) in the containment sump is reviewed. The areas of the containment which are exposed to the spray but are without direct drains to the recirculation sump (such as the refueling cavity) are considered. The reviewer confirms that the required sump concentrations are achieved within the appropriate time intervals. In addition, the long-term sump pH requirements (see Standard Review Plan 6.1.3) are considered for systems with low pH values.

The equilibrium partitioning of iodine between the sump liquid and the containment atmosphere is examined for the extremes of the additive concentrations determined above, in combination with the range of temperatures possible in the containment atmosphere and the sump solution. The minimum iodine partition coefficient determined for these conditions forms the basis of the ultimate iodine decontamination factor allowed in the staff's analysis, described below. (See Ref. 3 for a theoretical examination of iodine partition coefficients.)

f. Storage of Additives

The design of the additive storage tanks is reviewed to establish whether heat tracing is required to prevent freezing or precipitation in the tanks. The

reviewer determines whether an inert cover gas is provided for the tanks to prevent reactions of the additive with air, such as the formation of sodium carbonate by the reaction of sodium hydroxide and carbon dioxide. Alternatively, the reviewer verifies by a conservative analysis that an inert cover gas is not required.

g. Single Failure

The system schematics are reviewed by inspection, postulating single failures of any active component in the system, including inadvertent operation of valves that are not locked open. The review is performed with respect to the iodine removal function, considering conditions that could result in too fast as well as too slow an additive injection.

2. Testing

At the construction permit stage, the containment spray concept and the proposed tests of the system are reviewed to confirm the feasibility of verifying the design functions by appropriate testing. At the operating license stage, the proposed tests of the system and its components are reviewed to verify that the tests will demonstrate that the system, as installed, is capable of performing, within the bounds established in the description and evaluation of the system, all functions essential for effective iodine removal following postulated accidents.

3. Technical Specifications

The technical specifications are reviewed to verify that the system, as designed, is capable of meeting the design requirements and that it remains in a state of readiness whenever the reactor is critical.

a. Limiting Conditions for Operation

The LCO's should require the operability of the containment spray pumps, all associated valves and piping, the spray additive tanks including the appropriate quantity of additives, and any metering pumps or mixing devices.

b. Tests

Pre-operational testing of the system, including the additive tanks, pumps, (if any), piping, and valves is required, as discussed above. In particular, the pre-operational testing should verify that the system, as installed, is capable of delivering a well-mixed solution containing all additives with concentrations falling within the design margins assumed in the dose analyses of Chapter 15 of the SAR.

Periodic testing and exercising of all active components should include the spray pumps, metering pumps (if any), and valves. Confirmation that passive components, such as all essential spray and spray additive piping, and any passive mixing devices are free of obstructions should be made periodically. The contents of the spray additive tanks should be sampled and analyzed periodically to verify that the concentrations are within the established limits, that no concentration gradients exist, and that no precipitates have formed.

4. Evaluation

A calculation of the iodine removal effectiveness of the system is performed to establish the degree of iodine dose mitigation by the containment spray following the postulated accident. The mathematical model used for this calculation is designed to reflect the conclusions reached in the preceding steps of the review. The parameters determined include:

a. The maximum elemental iodine decontamination factor (DF) for the containment atmosphere achieved by the spray system, as determined from the equilibrium iodine partition coefficient. The DF is at present limited to a maximum value of 100 for sodium hydroxide and hydrazine systems and a value of 1000 for systems using appropriately buffered thiosulfate additive.

b. The removal of iodine from the containment atmosphere is considered a first-order removal process. The removal coefficient (lambda) for this process is determined for each of the sprayed regions of the containment, or, alternatively, an equivalent lambda for the entire free volume of the containment is determined. (See reference 4 for an acceptable model for the calculation of lambda.) The removal constant determined by this method is at present limited to a maximum value of 10 per hour for elemental iodine when used in conjunction with the instantaneous containment plate-out factor of 2 implied in Regulatory Guides 1.3 and 1.4.

IV. EVALUATION FINDINGS

If the AAB finds, on completion of the review, that the containment spray and spray additive system (if any) is effective for iodine removal, the following can be reported in the staff's safety evaluation report (SER):

"The concept upon which the proposed system is based has been demonstrated to be effective for iodine absorption and retention under post-accident conditions.

"The proposed system design is an acceptable application of this concept. The proposed pre-operational tests, post-operational testing and surveillance, and proposed limiting conditions of operation for the spray system provide adequate assurance that the iodine scrubbing function of the containment spray system will meet or exceed the effectiveness assumed in the accident evaluation."

In addition, the staff's evaluation of the iodine removal effectiveness of the containment spray system includes the following parameters, which are also used in the thyroid dose calculations of a postulated loss-of-coolant accident:

a. A first-order removal constant of λ_1 (1/hr) for elemental iodine, λ_2 (1/hr) for organic iodine, and λ_3 (1/hr) for particulate iodine.

b. An effective volume for the spray of V (ft^3).

c. A maximum decontamination factor (DF) for elemental iodine.

V. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

2. 10 CFR Part 50, Appendix A, General Design Criteria 41, 42, and 43.

3. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems - Part IV. Calculation of Iodine Partition Coefficients," ORNL-TM-2412 Part IV, Oak Ridge National Laboratory (1970).

4. A. K. Postma and W. F. Pasedag, "Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containments," WASH-1329, U.S. Atomic Energy Commission, June 1974.

5. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2.

6. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2.

7. General references for this SRP are listed in the following sections of the AAB "Bibliography for Filters, Sprays, and Iodine," maintained in the AAB office:

   Section IV, Design Considerations of Reactor Containment Spray Systems
   Section V, General Fission Product Behavior
   Section VI, Hydrazine Sprays
   Section IX, Iodine Species and Physical Properties
   Section XIII, Properties of Iodine-Water Systems
   Section XV, Spray Removal Systems
   Section XVI, Progress Reports - BMI
   Section XVII, Progress Reports - ORNL
   Section XVIII, Progress Reports - PNL
   Section XIX, Conferences

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STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.5.3 FISSION PRODUCT CONTROL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)
Auxiliary Systems Branch (ASB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The descriptions of the primary and secondary containments and of the containment penetrations are reviewed to (a) provide a basis for developing the mathematical model for design basis accident (DBA) dose computations, (b) verify that the values of certain key parameters are within pre-established limits, (c) confirm the applicability of important modeling assumptions, and (d) verify the functional capability of the secondary containment ventilation systems. The parameters which must be established and the systems whose functions must be reviewed or understood by the reviewer are outlined below. Many of these areas are the responsibility of other branches and are reviewed by the AAB to provide a general knowledge of the containment systems and their operation following a loss-of-coolant accident (LOCA).

1. Primary Containment Design

The following areas are reviewed:

a. Containment type, e.g., free-standing steel shell, reinforced steel-lined concrete, as described in Sections 3.8.1 or 3.8.2 and 6.2.1 of the applicant's safety analysis report (SAR). The containment type should be known so that the reviewer understands the degree to which positive pressure periods in the secondary containment may be affected by design basis accident heat loads on the primary containment. The need for containment vacuum relief valves may also be indicated by containment type. The CSB has responsibility for evaluating the pressure transient of the primary containment and for reviewing the vacuum relief valve design, where appropriate.

b. Pressure suppression devices, e.g., sprays, subatmospheric operation, suppression pool, ice condenser, as described in Sections 6.2.1 and 6.2.2 of the SAR. The existence and operation of pressure suppression devices should be determined since their existence and performance control peak containment pressure and

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containment leakage rate. The CSB is responsible for evaluating the peak containment pressure and containment leakage rate.

c. Fission product cleanup, e.g., sprays with chemical additives, internal ESF filter systems, ice condenser, as described in SAR Sections 6.5.1, 6.5.2 and 6.5.4. Knowledge of these systems is necessary for modeling the system for dose calculations.

d. General design characteristics, e.g., design leakage rate, free volume, fan flow rate across operating floor (ice condenser), peak containment pressure, time into a design basis accident for initiation and rate of hydrogen purge through purge system when exhausted into the secondary containment system or the environment through ESF filters. (See SAR Sections 9.4, 6.2.5, and Tables 6-1 through 6-4 as appropriate.) Some of these parameters are required for the dose calculations; others are required in establishing the model to be used.

The type of hydrogen removal system (recombiners, hydrogen purge or other) and some of the parameters needed for modeling, e.g., hydrogen purge initiation time and purge rate are interfaces with CSB, as detailed in Section III of this Standard Review Plan (SRP). Verification of other design data may require interfaces with CSB, ASB, MEB, SEB, or ETSB as noted in Section III.

2. Secondary Containment Design

The following areas are reviewed:

a. Containment type, e.g., metal siding, reinforced concrete. (See SAR Section 3.8.4.) The type of secondary containment structure may indicate the effect of varying wind speed (possible exfiltration) and the probable leak tightness of the secondary containment. The SEB has responsibility for reviewing the structural design of the containment. Leak tightness and leakage testing are the responsibility of the CSB.

b. Physical layout, e.g., volume completely surrounding primary containment, auxiliary building regions treated, main steam tunnel treated (in boiling water reactors), main steam line leakage control system provided (BWRs), drawings or plan views defining secondary containment boundary, clarification of which regions are treated by cleanup systems. (See SAR Sections 6.2.3, 6.5.3, and 9.3.) Knowledge of what regions are treated as part of the secondary containment is essential to establish the mathematical model for dose calculations.

c. Fission product removal or holdup system design, e.g., regions treated by each system, piping and instrumentation drawings of each system and its operation, fan flow rates, recirculation rate, filter locations and efficiencies, system redundancy, actuation signals, time to reduce region pressures below atmospheric, placement of ducting. (See SAR Sections 6.2.3, 6.5.1, and 6.5.3.) The reviewer

is responsible for determining that each system can perform its functions as claimed to reduce fission product release following a postulated design basis accident. Information on fission product removal systems may be provided by AAB (spray system) or by the ETSB (filter system). Knowledge of these systems is necessary for modeling the system for the dose calculation. CSB has responsibility for evaluating the pressure transient in the secondary containment to verify secondary containment region pressures following a design basis accident and for reviewing bypass leakage paths. MEB has responsibility for evaluating the structural design of the ventilation system.

d. General design characteristics, e.g., negative pressure maintenance during normal operation, free volumes of regions, and leakage rates. (See SAR Sections 6.2.3, 6.5.3, and 9.4.) Knowledge of these parameters is also necessary for developing the mathematical model. The ASB has responsibility for evaluating systems which maintain negative pressure in secondary containment regions during normal operation. The CSB has responsibility for evaluating secondary containment leakage rates.

II. ACCEPTANCE CRITERIA

In establishing the model to be used for estimating the radiological consequences of a design basis loss-of-coolant accident and determining the acceptability of the secondary containment ventilation systems, the following acceptance criteria are used by the AAB.

1. Primary Containment

Primary containment design leakage rates for which credit is given should not be less than 0.1%/day due to difficulties in measuring lower leakage rates. Since the leak rate is a major contributor to the LOCA dose, the upper bound on the leak rate is in part determined by 10 CFR Part 100 guideline calculations.

2. Secondary Containment

To be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and its volume should be held at a minimum negative pressure differential of 0.25 inch (water), when compared with adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to assure site boundary exposures less than those calculated for the design basis accidents even if exfiltration occurs. (For a very leaky secondary containment, the CSB requests the AAB to perform a special exfiltration analysis.) Metal siding structures are acceptable if they can meet all leakage test requirements under varying wind conditions.

Other criteria include specifications for intake and return headers on recirculation systems. These should be placed as far away from each other as is practical. The return header should provide a wide distribution over the confinement volume. The purpose of this placement is to assure some degree of mixing of the return flow in the secondary containment volume before it is again drawn into the system intake.

With judicious placement, up to 50% mixing may be assumed, but a claim for greater than 50% mixing must be supported by the applicant to the satisfaction of the staff. Spacing between intake and return headers is reviewed on a case-by-case basis. Adjustments in the mixing fraction to less than 50% may be indicated by some designs. Past practice has been to allow mixing in 50% of the volume between (and within 10 or 20 feet of) the inlet and outlet headers if both have distributed openings or if one has distributed openings and the other is at the top of the containment.

3. Partial Dual Containment

Partial dual containments must meet the same basic requirements as those for secondary containments in order to be given credit for fission product holdup and removal.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP section(s) corresponding to those branches.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The purpose of the review of containment systems is to define a model to be used in DBA (specifically, the LOCA) dose calculations, to check that the values of certain key parameters are within pre-established limits, to confirm the correctness of important modeling assumptions, and to verify the functional capability of the primary and/or secondary containment ventilation systems. Specific system design areas may not be reviewed in detail (filters, sprays, leakage rates, etc.), but the reviewer is responsible for reviewing all related ventilation systems and for selecting a representative dose model for DBA calculations. Therefore, the reviewer covers various areas (containment design, positive pressure periods, filters, etc.) for continuity rather than detail. Digital computer codes (Ref. 1) are used to perform the dose calculations.

All statements referring to "operation" in the following discussion mean operation following a postulated design basis LOCA. Normal operation is so identified.

Where a review area is not the primary responsibility of the AAB, appropriate acceptance criteria in applicable SRPs are used by the responsible branch and the AAB is informed where inadequacies are identified so that appropriate modifications of the model may be made. These areas include:

- Primary containment leakage rate, bypass leakage, and testing of these (CSB).
- Hydrogen purge systems (CSB).
- Secondary containment vacuum maintenance systems (normal operation) (ASB).

- Secondary containment pressure response (post-accident) (CSB).
- Containment isolation (CSB).
- Containment isolation (CSB).
- Structural design of containments (SEB) and systems (MEB).
- Engineered safety feature filter systems (ETSB).

1. Primary Containment Design

   a. The primary containment design is studied to familiarize the reviewer with the overall construction and anticipated performance capability of the primary containment. Certain parameters, such as design leakage rate, containment free volume, the existence of internal fission product cleanup systems, should be noted for later use. (See example of worksheet, Table 6.5.3-1.) The performance capability of the internal fission product cleanup systems (if any) should be verified. (See SAR Sections 6.5.1, 6.5.2, and 6.5.4.)

   b. The curve indicating containment pressure versus time following the accident should be studied. Historically, pressurized water reactor (PWR) containment design leakage rates have been reduced by a factor of two at one day into the accident (see Ref. 2), whereas, boiling water reactor (BWR) containment design leakage rates were assumed to be constant for all time periods following the accident. (See Ref 3.) The reviewer should verify with CSB that these modeling assumptions are valid for each case reviewed. For those containments designed to reach subatmospheric pressure at some time less than 30 days after the accident, the CSB verifies the time required to reach subatmospheric pressure.

      Performance of the hydrogen purge system is reviewed for the purpose of modeling the purge dose calculation where the purge system is relied upon as the primary means of controlling hydrogen. On some systems, the hydrogen purge lines are vented to the recirculation return line of the secondary containment ventilation system. Initiation time for the hydrogen purge and the purge rate is obtained from the CSB.

2. Secondary Containment Design

   a. The secondary containment design is reviewed to determine how it should be modeled for the dose calculations. The ability of the structures to withstand the safe shutdown earthquake or to meet the tornado criteria is the responsibility of the SEB, but the reviewer checks the applicant's SAR to determine what tornado missile and wind criteria the structures are designed to meet. The reviewer also ascertains that the applicant has considered the question of potential exfiltration from regions of the secondary containment under varying wind conditions, especially if the structure has a leakage rate greater than 100%/day. The anticipated leakage rate from each region is noted (see example of worksheet,

Table 6.5.3-2), and special attention paid to the accuracy of the proposed leakage testing if the leakage rates are less than 10% per day. (No facility reviewed to date has a proposed secondary containment leakage rate of less than 10% per day. Experience indicates that 10% per day may be difficult to achieve in actual practice.)

b. The boundary of the secondary containment is determined. Usually, the secondary containment boundary is composed of more than one region, e.g., a shield building (concrete) or enclosure building (metal siding) around the primary containment and all or parts (emergency core cooling pump rooms, etc.) of the auxiliary building. These regions may be treated by one or more ventilation systems.

c. For PWR containments and BWR MARK III containments, the annular region between the shield building or enclosure building and the primary containment may be held at a negative pressure relative to adjacent areas by a vacuum exhaust system during normal operation. Since this system is used during normal operation, it may appear in the SAR under auxiliary systems. The exhaust system may also treat the auxiliary building regions which are part of the secondary containment; but if these regions are maintained at a negative pressure during normal operation, it is most likely done with the auxiliary building ventilation system. Both the vacuum exhaust and auxiliary building ventilation systems fall under the purview of the ASB. The systems' ability to maintain negative pressures of sufficient margin under varying wind conditions and operational modes prior to a design basis accident is verified by the ASB. The AAB reviewer is responsible for reviewing the design of systems maintaining negative pressure following a design basis accident. If an adequate negative differential pressure margin (0.25 inch water gauge) achieved within 60 seconds from the time the accident happens, then no positive pressure time period need be assumed in the dose model. All positive pressure periods at any time in the secondary containment regions are treated as direct outleakage periods following an accident, and no credit is given for filters or recirculation systems. The CSB verifies the positive pressure periods. The large reactor buildings around older BWR containments are usually maintained at a negative pressure during normal operation, and the dose model used for these cases has not assumed any positive pressure period.

d. The exhaust systems used to maintain the negative pressure differential following the accident should be sized to meet the negative pressure criterion for the inleakage rate and the conservatively calculated heat load for the regions treated by each, and analyses to this effect should be presented by the applicant. The pressure response analyses are reviewed by the CSB. The functional capability of the filter design associated with the exhaust system is reviewed by the ETSB under Standard Review Plan (SRP) 6.5.1. The reviewer should establish that the ESF filter systems are being reviewed by the ETSB. The exhaust systems may be one of several designs. Common designs are:

(1) Straight exhaust through charcoal and HEPA filters. Primary containment leakage to these regions is assumed to go directly to the filter with no mixing or holdup in the region being filtered.

(2) Recirculation system with split in flow (some exhausted through filters and some recirculated to the region being treated). Primary containment leakage to the region being treated is assumed to be directly to the intake of the recirculation fan. There, a fraction of it (the ratio of exhaust to total flow) is exhausted through the filters; the balance is then assumed to return to the region being treated. The placement of the system intake and return headers is examined to determine that return flow from the fans does not have a direct path to the intake again. Credit for mixing in 50% of the region is given for fission products returned by the recirculation system to the secondary volume if the header placement is satisfactory.

(3) Other variations on the recirculation system are (a) filters in the recirculation line, (b) filters in both the recirculation line and the exhaust line, and (c) high exhaust flow to reduce the negative pressure to several inches water gauge, and then no exhaust with recirculation only for some time period.

The sizing of the system fans for the volumes they are maintaining at a negative pressure may be critical in determining the ratio of exhuast flow to recirculation flow. Past history shows secondary containment structures are considerably more leaky than applicants anticipated (2 to 5 times as great as anticipated), and fan exhaust flows have been increased after testing to account for this. (When identical flow rates are predicted for two volumes which differ by a factor of 10 or more, it is difficult to believe that the negative pressure differential will be the same for both volumes.) The flow rates, negative pressure differential, and volumes are noted and the appropriate AAB reviewer and CSB reviewer (pressure response only) consulted for verification before performing dose calculations.

The systems should be reviewed to determine volumes treated, system operation, fan flow rates, and filter efficiencies. All the applicant's claims should be verified by appropriate staff members as noted on Table 6.5.3-2. Leakage fractions from the primary containment to each volume should be identified and stated in the technical specifications. Completeness of information, adequacy of technical specifications and testing methods, and the adequacy and maintenance of the integrity of the secondary containment negative pressure considering failures of nonseismic piping or ducting are verified by the CSB.

IV. EVALUATION FINDINGS

The reviewer defines a dose model for the LOCA dose calculations and prepares a table of all the data for the primary and secondary containments to be used in the calculation. The recommended form for tabulation is given in Table 6.5.3-3. This table should also include the information needed to model hydrogen purge dose calculations. In addition, the reviewer verifies that sufficient information has been provided and that the review and calculations

support conclusions of the following type, to be included in the staff's safety evaluation report:

"The fission product control systems include all structures, ducting, valves, and fans which are used to control leakage of fission products following a postulated design basis accident. The scope of review of these systems included piping and instrumentation diagrams and general arrangement diagrams showing flow in the fission product control systems and areas treated by each system, and descriptive information about each system. The review has included the applicant's proposed design criteria and design bases for each system and the applicant's analysis of the adequacy of those criteria and bases. The applicant's analyses of the manner in which the designs of the fission product control systems conform to the proposed design criteria have also been reviewed.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the fission product control systems and necessary auxiliary supporting systems to the Commission's regulations, and to Regulatory Guide 1.3 (or 1.4), staff technical positions, and industry standards.

"The staff concludes that the designs of the fission product control systems conform to all applicable regulations, guides, staff positions, and industry standards, and are acceptable."

V. REFERENCES

1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2.

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2.

Table 6.5.3-1

Example Worksheet:
Primary Containment Information

Data Description	Parameter Value	Staff Verification
Type of Structure		SEB
Primary Containment Design Leak Rate		CSB
Bypass Leakage Fraction to Volumes		CSB
1.		
2.		
3.		
Primary Containment Free Volume		CSB
Primary Containment Subatmospheric Operation		CSB
Primary Containment Internal Fission Product Removal Systems:		AAB
Ice Condenser		
Spray System		
Filter System		
Other		
H_2 Purge Mode (e.g., direct, to recirculation systems, to annulus)		CSB
Purge Initiation Time		
Purge Rate		
Primary Containment Purge:		CSB
Used During Normal Operation		
Valve Arrangement		

Table 6.5.3-2

Example Worksheet:
Secondary Containment Information

Data Description	Parameter Value	Staff Verification
For each Secondary Containment Region:		
Type of Structure		SEB
Free Volume		CSB
Mixing Fraction		AAB
Design Leak Rate		CSB
Annulus Width (where applicable)		CSB
For each Ventilation System:		
Total Recirculation Flow		AAB
Exhaust Flow		AAB
Filter Placement		AAB
Filter Efficiencies		ETSB
Header Placement		AAB
Time Sequence for Operation Following an Accident or		CSB
Operation of System Prior to an Accident if Used During Normal Operation		ASB

Table 6.5.3-3

Evaluation Findings

Primary Containment Leak Rate

Primary Containment Free Volume

Primary Containment Internal Fission Product Removal System

Primary Containment Subatmospheric Operation

Primary Containment Leakage Paths

Secondary Containment Free Volume

Secondary Containment Total System Flow

Secondary Containment Exhaust Flow

Secondary Containment Mixing Fraction

Secondary Containment Filter Efficiencies

Time Sequence for Operation of Fission Product Removal or Holdup Systems in Total Containment System Following a Postulated Accident

H_2 Purge:

- Initiation Time
- Purge Rate
- Purge Model

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SECTION 6.5.4 ICE CONDENSER AS A FISSION PRODUCT CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) are reviewed:

1. Fission Product Removal Requirement for the Ice Condenser System

   Sections of the SAR related to accident analysis, dose calculations, and fission product removal and control are reviewed to establish whether fission product scrubbing of the containment atmosphere is required for mitigation of offsite doses following a postulated accident. This review usually covers SAR Sections 6.2, 6.5.4 and 15.2.X.X.

2. Design Bases

   The design bases for the fission product removal function of the ice condenser system are reviewed to determine whether they are consistent with the requirements placed upon this system by the assumptions made in the accident evaluations of SAR Chapter 15.

3. System Design

   The descriptive information concerning the portions of the ice condenser system design important to its fission product removal function is reviewed to familiarize the reviewer with the design and post-accident functioning of the ice condenser. This includes:

   a. The basic design concept, the systems, subsystems, and support systems required to carry out the fission product cleanup function of the ice condenser.

   b. Descriptive information and figures from SAR Section 6.2, as related to:

      (1) The time required to establish a steady flow of predictable magnitude of an air-steam-iodine mixture through the ice beds.

      (2) The time of melt-out of the ice beds.

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4. Testing and Inspections

The details of the applicant's proposed preoperational test to be performed for system verification and operational tests and inspections to verify the continued status of readiness of the iodine removal capacity of the ice condenser systems are reviewed.

5. Technical Specifications

At the operating license stage, Sections 3 and 4 of SAR Chapter 16 are reviewed to establish surveillance requirements for the sodium hydroxide concentrations in the ice.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the fission product cleanup function of the ice condenser system are:

1. Ice Alkalinity

The ice condenser system is acceptable for elemental iodine removal if the ice contains a quantity of sodium hydroxide sufficient to assure that the water solution from ice melting has a pH of at least 9.0 but not more than 11.0.

2. Duration of Iodine Scrubbing Function

The ice condenser is assumed to be effective for iodine removal only during that period following an assumed accident when a steady flow of predictable magnitude of the air-steam-iodine mixture has been established. Steady flow is assumed to commence with the operation of the post-accident mixing fans if time-independent iodine removal efficiency is assumed (i.e., 30%/pass).

3. Tests and Inspections

Preoperational and inservice tests should assure that the proper ice alkalinity is maintained. Other inspections associated with the pressure suppression function will assure the adequacy of ice quantity and geometry.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this SRP section as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The first step in the review of ice condenser fission product removal is to determine whether the ice condenser system is used for accident dose mitigation purposes. Chapter 15 of the SAR is reviewed to determine whether a dose reduction credit was assumed for the ice condenser. If no fission product removal credit is assumed in the accident analysis, no further review is required.

If the ice condenser system is used for iodine removal, the iodine removal effectiveness of the ice condenser system is reviewed. The review includes the following:

1. System Design and Evaluation

   a. Ice Chemistry

   The chemistry of the ice is usually modified to include sodium hydroxide in order to improve the iodine scrubbing effectiveness of the ice condenser system. If the concentration of the sodium hydroxide is such that the ice, after melting but prior to any dilution meets the pH requirements (i.e., $11 \geq pH \geq 9$) stated in the acceptance criteria of this SRP section, the system is considered effective for elemental iodine removal. For ice condenser systems similar to those of the D. C. Cook and Sequoyah plants (with a steady-state flow rate of approximately 40,000 cfm) an efficiency of 30% per pass for elemental iodine is assigned. The system is considered ineffective for organic and particulate iodine removal. For designs different from those of D. C. Cook or Sequoyah, reconsideration of the system efficiency is required. References 4 and 5 should be reviewed to aid in reviewing the applicant's analyses. Similarly, Reference 4 should be reviewed if it is proposed that time-dependent removal efficiencies be used. Any dependencies based on time-varying air-steam mixtures or flow rates should conservatively account for factors that affect such dependencies (e.g., fan capacity, fan activation time, natural circulation rates).

   b. Iodine Scrubbing Function

   It is not feasible to specify the exact time of the fission product release following a postulated loss-of-coolant accident. In addition, the flow rates and air-steam fractions of the flow through the ice condenser vary significantly during and immediately following the accident. For dose calculation purposes, therefore, the following conservative assumptions are made:

   (1) If a 30% iodine removal efficiency is used, the iodine removal effectiveness of the ice condenser commences with the establishment of a steady-state air-steam flow by the air-steam return fans. (A single failure of one of the fans is assumed.)

   (2) The initial concentration of iodine is assumed uniform throughout the entire containment.

   (3) The effectiveness of the ice condenser as an iodine removal system is assumed to cease with the melt-out of the first ice bed.

   c. Upper Compartment Spray Credit

   Plants designed with an upper compartment spray system may claim credit for such. SRP Section 6.5.2 is used to review such spray systems.

d. Evaluation

The air-steam fan flow rate is used with the above assumptions in modeling fission product behavior for the loss-of-coolant accident (see Appendix A to SRP Section 15.6.5).

2. Technical Specifications

The technical specifications are reviewed to assure that they require periodic inspection and sampling of the ice in order to confirm the continued state of readiness of the system, i.e., the system meets the chemistry requirements specified in the acceptance criteria of this SRP section.

V. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's Safety Evaluation Report:

"We have reviewed the fission product scrubbing function of the ice condenser and conclude that the addition of sodium hydroxide to the ice, as proposed by the applicant, will reduce the elemental iodine concentration of the steam-air mixture flowing through the ice beds following a loss-of-coolant accident. We estimate an elemental iodine removal efficiency of __% per pass during the time period starting at __ minutes after the accident and ending at ___ minutes. The applicant's proposed program for preoperational and periodic surveillance tests will assure a continued state of readiness for the ice condenser iodine removal function."

V. REFERENCES

1. (CONF-700608-16) Iodine Removal in the Ice Condenser System; Malinowski, D. D.; Picone, L. F. (Westinghouse Electric Corporation, Pittsburgh, PA, Nuclear Energy Systems Division); 1970.

2. Iodine Removal in the Ice Condenser System; Malinowski, D. D.; Picone, L. F. (Nuclear Energy Systems, Westinghouse Electric Corporation, Pittsburgh, PA); Nuclear Technology 1971, 10(4), 428-35.

3. Basic Properties of Ice as Related to the Performance of Ice Condensers, WASH-1232; Soldano, B. A. (U.S. Atomic Energy Commission, Washington, DC); September 1972.

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5. Review of Topical Report WCAP-7426; memo from H. R. Denton to R. C. DeYoung, U.S. Atomic Energy Commission, November 24, 1972.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

General Design Criterion 36, "Inspection of Emergency Core Cooling System;" Criterion 39, "Inspection of Containment Heat Removal System;" Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems;" and Criterion 45, "Inspection of Cooling Water System," of Appendix A to 10 CFR Part 50 require that the subject systems be designed to permit appropriate periodic inspection of important component parts to assure system integrity and capability.

The following areas relating to the inservice inspection (ISI) program for AEC Quality Group B and C (ASME Boiler and Pressure Vessel Code, Section III, Code Class 2 and 3) components are reviewed:

1. Components Subject to Examination

The descriptive information in the applicant's safety analysis report (SAR) is reviewed to establish that all the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Subarticle NA-2140, Class 2 and Class 3 components are included in the ISI program. The Reactor Systems Branch verifies that the system classifications as Code Class 2 and 3 agree with Subarticle NA-2110 of Section III and with the definitions of the general design criteria.

2. Accessibility

The descriptive information, including drawings, is reviewed by the Materials Engineering Branch to establish that the Code Section XI, Subarticle IWA-1500, provisions for system accessibility are included in the applicant's layout and design of these systems.

3. Examination Techniques and Procedures

The required examination techniques and procedures, including the requirements of Subarticles IWC-2600 and IWD-2600 of Section XI, are reviewed for their conformance to Subarticle IWA-2200 of Section XI of the Code.

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

4. Inspection Intervals

The required examinations and inspections listed in the SAR are reviewed and compared to Table IWC-2600 and Subarticle IWD-2600 of Section XI to verify that they will be performed within the designated inspection interval and comply with subarticles IWC-2400 and IWD-2400 of Section XI to verify that they will be performed within the designated inspection interval and comply with Subarticles IWC-2400 and IWD-2400 of Section XI.

5. Examination Categories and Requirements

The technical specifications are compared to Table IWC-2520 and Subarticle IWD-2600 of Section XI to verify that the examination categories and ISI requirements of each category are in agreement.

6. Evaluation of Examination Results

The information concerning repair procedures is reviewed for compliance with Articles IWC-4000 and IWD-4000 of Section XI. Because Articles IWC-3000 and IWD-3000 are still in course of preparation, as an interim step, evaluation of examination results for Class 2 and 3 components is reviewed for compliance with Article IWB-3000 for Class 1 components.

7. System Pressure Tests

The pressure test program is reviewed for compliance with Subarticles IWC-5200 and IWD-5200 of Section XI to establish that leakage and signs of structural distress are inspected for on a periodic basis. In addition, for Class 2 components, a review is performed to determine that all applicable systems and components are pressure tested at acceptable combinations of temperature and pressure.

8. Augmented ISI to Protect Against Postulated Piping Failures

The augmented inservice inspection program to provide assurance against postulated piping failures of high energy fluid systems between containment isolation valves is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Components Subject to Inspection

The applicant's definition of Code Class 2 and 3 components and systems subjects to an ISI program is acceptable if in agreement with the definitions of Code Section III, Subarticle NA-2140, and Section XI, Table IWC-2600 and Subarticle IWD-2600. The interpretation of code classifications by the applicant is subject to review by the Reactor Systems Branch for compliance with safety criteria pertaining to component classification. (Refer to NA-2110 of Section III.)

2. Accessibility

The design and arrangement of Class 2 and 3 systems are acceptable if the applicant includes allowances for adequate clearances to conduct the examinations specified in IWC-2600 and IWD-2600 at the frequency specified by IWC-2400 and IWD-2400. Special

design considerations are given to those systems that are intended to be examined during normal reactor operation.

3. Examinatin Techniques and Procedures

Since acceptance criteria for the required examination techniques and procedures are identical to those for the reactor coolant pressure boundary (RCPB), Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing" (Ref. 3), Section II.3, "Examination Techniques and Procedures," is incorporated by reference in this plan.

4. Inspection Intervals

The inservice inspection program schedule given in the SAR is acceptable if the required examinations are completed during each ten-year interval, hereafter designated as the inspection interval, and as required by Subarticles IWC-2400 and IWD-2400 of Section XI.

5. Examination Categories and Requirements

The ISI examination categories and requirements for Class 2 and 3 components established by the SAR are acceptable if in agreement with the following criteria of Section XI: IWC-2520, IWC-2600, and IWD-2600. The areas subject to examination and the extent of examination for the Class 2 components shall comply with the requirements specified under the examination categories of Table IWC-2520.

6. Evaluation of Examinations Results

Articles IWC-3000 and IWD-3000 of Section XI concerning evaluation of examination results are still in preparation. The applicant's evaluation of examination results should be consistent with that for Code Class I components. Thus, Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing," Section II.6, "Evaluation of Examination Results," is considered acceptable (Ref. 3).

7. System Pressure Tests

The SAR program for Class 2 and 3 system pressure testing is acceptable if it meets the following criteria of IWC-5000 and IWD-5000 of Section XI:

a. Class 2 Systems

(1) The system hydrostatic test pressure shall be at least 1.25 times the system design pressure (P_D) and conducted at a test temperature not less than 100°F except as may be required to meet the test temperature requirements of IWA-5230.

(2) When system hydrostatic testing is required to be conducted at temperatures above 100°F in order to meet the fracture toughness criteria applicable to ferritic materials of which the system components are constructed, the test pressure may be reduced in accordance with the table given in IWC-5220.

(3) For components that are not required to function during reactor operation, the system test pressure shall not be less than 100 percent of the pressure developed during the conduct of a periodic system performance test.

(4) Open-ended portions of a system (e.g., suction line from a storage tank, or discharge line of a containment spray header) extending to the first shutoff valve may be exempted from the test requirements of IWC-2510.

b. Class 3 Systems

(1) The system test pressure shall be at least 1.10 times the system design pressure.

(2) Open-ended portions of a system (e.g., suction line from a storage tank) extending to the first shutoff valve may be exempted from the test requirements of IWD-5200.

c. Storage Tanks

The nominal hydrostatic pressure developed with the tank filled to its design capacity shall be acceptable as the system test pressure.

8. Augmented ISI to Protect Against Postulated Piping Failures

High energy fluid system piping between containment isolation valves should receive an augmented ISI as follows:

a. Protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the Code, Section XI, Division 1.

b. For those portions of high energy fluid system piping between containment isolation valves, the extent of inservice examination completed during each inspection interval (IWA-2400, ASME Code Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

c. For those portions of high energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

d. The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Tables IWC-2520.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review the following review procedure is followed:

1. Components Subject to Inspection

The applicant's components and system classifications under Class 2 and 3 are reviewed for agreement with Section II.1 of this plan. The interpretation of code classifications is the responsibility of the Reactor Systems Branch, should a discrepancy occur between the SAR and Section II.1.

The applicant's classification of Class 3 systems is reviewed for agreement with Section II.1 of this plan. Any safety-related, fluid-carrying components not included in Class 1 or Class 2 and not a part of the containment structure are included in Class 3.

2. Accessibility

The design and arrangement of Class 2 and 3 systems are reviewed in terms of accessibility for ISI to establish that the design meets the requirements of Section II.2 of this plan. No remote inspection program is required for Code Class 2 or 3 components.

3. Examination Techniques and Procedures

The reviewer verifies that the examination techniques as described by the SAR are the same as those specified in Section II.3 of this plan.

4. Inspection Intervals

The inservice inspection program for Class 2 and 3 components in the plant technical specifications is reviewed to establish that each area and component in the program is inspected on a schedule in agreement with Section II.4 of this plan.

5. Examination Categories and Requirements

The examination categories and parallel inspection requirements described or tabulated in the technical specifications are reviewed to establish that they are in agreement with Section II.5 of this plan. The technical specification table or description is acceptable if it follows Tables IWC-2520 and IWC-2600 in terms of headings and arrangement for Class 2 components and IWD-2600 for Class 3 components.

6. Evaluation of Examination Results

The reviewer verifies that the evaluation of examination results described in the SAR is in accord with Section II.6 of this plan.

7. System Pressure Test

The system pressure test program is acceptable if it meets the criteria of Section II.7 of this plan.

8. Augmented ISI to Protect Against Postulated Piping Failures

The reviewer verifies that the augmented inservice inspection program as described in the SAR meets the acceptance criteria identified in Section II.8 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"To ensure that no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones are inspected prior to reactor startup and periodically throughout the life of the plant. In addition, Code Class 2 and 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

"Examples of Code Class 2 systems are: residual heat removal systems, portions of chemical and volume control systems (in PWR plants), portions of control rod drive systems (in BWR Plants), and engineered safety features not part of Code Class 1 systems. Examples of Code Class 3 systems are: component cooling water systems, and portions of radwaste systems. All of these systems transport fluids. The applicant has stated that the design of Code Class 2 and 3 systems meets the requirements of ASME Code Section XI. Compliance with the inservice inspections required by the Code and staff technical positions constitutes and acceptable basis for satisfying applicable requirements of General Design Criteria 36, 39, 42, and 45."

V. REVERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System;" Criterion 39, "Inspection of Containment Heat Removal System;" Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems;" and Criterion 45, "Inspection of Cooling Water System."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Sections NA-2140 and NA-2110, and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

3. Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing."

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SECTION 6.7 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Containment Systems Branch (CSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Instrumentation and Control Systems Branch (ICSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

Direct cycle boiling water reactor (BWR) plants have redundant quick-acting isolation valves on each main steam line from the reactor to the turbine. In the event of a loss-of-coolant accident (LOCA), any leakage of contaminated steam through these valves is controlled by a leakage control system. The leakage control system must satisfy the requirements of General Design Criteria 2, 4, and 54.

The review of the main steam isolation valve leakage control system (MSIVLCS) covers the entire leakage control system including the source of the sealing medium, if any, and pumps, valves, and piping to the points of connection or interface with the main steam supply system. Emphasis is placed on the components of the leakage control system that are required to remain functional following a design basis LOCA.

1. ASB reviews the design of the MSIVLCS and essential subsystems to assure their ability to function following a postulated LOCA including the loss of offsite power. The system is reviewed to determine that:

   a. A malfunction or failure of an active component of the system, or loss of the source of sealing fluid, if any, will not reduce the functional performance of the system.

   b. The failure of non-seismic Category I equipment or components will not have an adverse effect on the ability of the system or components to function.

   c. The capability of the system to perform its intended safety function is maintained assuming a single active failure of a main steam line isolation valve.

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Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

2. The ASB also reviews the design of the leakage control system with respect to the following:

   a. The capability of the system to withstand the effects of the safe shutdown earthquake, including the source of sealing medium, if any.

   b. The capability of the system to control main steam isolation valve leakage and preserve containment integrity under design basis LOCA conditions, including loss of offsite power.

   c. The compatibility of initiation means and controls of the system with loading requirements on the emergency electrical buses, operator reaction times, and with actuation times available in view of the specified main steam isolation valve leakage limits.

   d. The requirements for interlocks to prevent inadvertent system operation.

   e. The capability of the system design to permit functional testing of components, controls, and actuation devices during power operations to the extent practicable and complete functional testing during plant shutdown.

   f. The capability of the system and main steam supply system components to withstand effects resulting from the use of a sealing medium, if any, such as thermal stresses, pressures associated with flashing, and thermal deformations, so that the structural integrity of the main steam lines and main steam isolation valves will not be affected and that any deformation of valve internals will not result in excessive leakage through the valves.

   g. The design provisions incorporated to prevent or treat main steam isolation valve stem packing leakage or other direct leakage.

   h. The instrumentation and control features necessary to accomplish the system function, including isolation of components of the system in the event of malfunctions.

   i. The use of applicable codes and standards and assignment of appropriate seismic and quality group classifications.

   j. The need for a third main steam shutoff valve in each main steam line upstream of the turbine stop valve to assure the safety function of the MSIVLCS.

3. The applicant's proposed technical specifications are reviewed at the operating license stage as they relate to areas covered in this SRP section.

Secondary review evaluations are performed by other branches and the results used by ASB to complete the overall evaluation of the system. The evaluations provided by other branches are as follows. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE). The MEB reviews the seismic qualification of components and confirms that components and piping are designed in accordance with applicable codes and standards. The MTEB verifies that inservice inspection requirements are met for system components and, upon request, verifies the compatibility of the materials of construction with service conditions. The ICSB and PSB determine the adequacy of the design, installation, inspection and testing of all electrical components (sensing, control, and power) required for proper operation. The CSB reviews the MSIVLCS to assure that no malfunction can adversely affect containment integrity.

II. ACCEPTANCE CRITERIA

Acceptability of the MSIVLCS, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. An additional basis for determining the acceptability of the MSIVLCS is the degree of similarity of the design with that of previously reviewed plants.

The design of the MSIVLCS is acceptable if the integrated system design is in accordance with the following criteria:

1. General Design Criterion 2, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, as related to structures housing the system and the system itself being capable of withstanding the effects of externally and internally generated missiles.

3. General Design Criterion 54, as related to the capability for leak detection, isolation, and performance testing for system piping penetrating containment.

4. Regulatory Guide 1.26, as related to the quality group classifications of components and systems.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Regulatory Guide 1.96, as related to the design of the MSIVLCS.

7. Regulatory Guide 1.102, as related to the protection of structures, systems, and components important to safety from the effects of flooding.

8. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles.

9. Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

10. Branch Technical Position ASB 3-2, as related to the classification of main steam components other than reactor coolant pressure boundary.

11. Branch Technical Position ASB 3-3, as related to the classification of BWR main steam and feedwater components other than reactor coolant pressure boundary.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria, design bases, and preliminary design meet the acceptance criteria given in subsection II. For the review of operation license (OL) applications, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design. The OL review includes a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements for system testing, minimum performance, and surveillance developed by the staff. The reviewer will select and emphasize material from this SRP section, as may be appropriate for a particular case.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

1. The information provided in the SAR pertaining to the design basis and design criteria, the system piping and instrumentation diagrams (P&IDs), and the system description are reviewed to determine that they clearly delineate the following:

   a. The method used to accomplish the main steam isolation valve leakage control function and the system components essential for operation following design basis LOCA conditions.

   b. Essential components of the leakage control system are correctly identified and are isolable from any non-essential portions of the system. The PID's are reviewed to verify that they clearly indicate the physical divisions between such portions and indicate any design classification changes. System drawings are reviewed to see that they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance requirements for the leakage control system isolation valves.

c. Essential components of the leakage control system, including the isolation valves separating any non-essential portions of the system, and the seal fluid source (if used) are classified seismic Category I and Quality Group A or B, as specified in Regulatory Guide 1.96. Component and system descriptions in the SAR that identify mechanical and performance characteristics are reviewed to verify that the above classifications have been included, and that the P&IDs indicate points of design classification changes.

d. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary design provisions to accomplish the testing program.

2. The reviewer determines that the safety function of the MSIVLCS will be maintained, as required, in the event of adverse environmental phenomena such as earthquakes. The reviewer uses engineering judgment, the results of failure modes and effects analyses, and the results of reviews performed under other SRP sections to determine that the failure of non-essential portions of the system or of other systems not designed to seismic Category I and located close to essential portions of the system, or of non-seismic Category I structures located close to essential portions of the system, will not preclude operation of the essential portions of the MSIVLCS. Reference to SAR sections describing site features, the general arrangement and layout drawings, and the tabulation of seismic design classifications for systems and structures will be necessary. Statements in the SAR that the above conditions are met are acceptable.

3. If the leakage control system is one using a fluid sealing medium:

a. The system design is reviewed to determine that the quantity of sealing fluid needed for an effective seal of the valves has been provided. Independent analyses, using the pump performance curves in the SAR, are made to assure that the design and the location of the pump and components are such as to maintain the appropriate net positive suction head (NPSH) requirements and provide a continuous supply of sealing fluid during the full course of an accident.

b. The system design is reviewed to determine that effects resulting from the sealing fluid, such as thermal stresses, pressures associated with flashing, thermal deformations, and other effects will not effect the structural integrity of the steam lines or the main steam isolation valves, or lead to excessive leakage of the valves. This portion of the review is done on a case-by-case basis. The ASB also accepts the system design if a statement in the SAR commits to performing calculations or functional testing to demonstrate that the above conditions are met.

4. The MSIVLCS is reviewed to verify that instrumentation, controls, and interlocks designed to standards appropriate for an engineered safety feature are provided to actuate the system in the event of a design basis LOCA, and to prevent inadvertent actuation. Interlocks to prevent inadvertent operation of the leakage control system that are actuated by signals from the reactor protection, engineered safety feature, or containment isolation systems are acceptable. A statement in the SAR that such instrumentation, controls, and interlocks will be provided is acceptable for construction permit (CP) review.

5. The system performance requirements, P&IDs, MSIVLCS drawings, and the results of failure modes and effects analyses are reviewed to assure that the system can function following a design basis LOCA assuming a concurrent single active failure, including the failure of a single main steam isolation valve to close. The reviewer evaluates the analyses presented in the SAR to assure the function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum requirements are met for each failure condition over the required time spans. For each case the design is acceptable if minimum system functional requirements are met. The reviewer also provides the Accident Analysis Branch with an estimate of the quantity of fluid processed by the MSIVLCS, for use in calculating radiological consequences of a LOCA.

6. The leakage control system design is reviewed to verify that valve stem packing leakage or other direct leakage from the main steam isolation valves or other components outside containment is prevented or controlled. Such leakage could bypass the leakage control system and result in untreated releases to the environment. The means for prevention or control need not be part of the leakage control system itself, but should meet the same design standards.

7. The leakage control system design is reviewed to determine if a third main steam line valve, located between the main steam isolation valve and the turbine stop valve, is required to assure that the MSIVLCS can perform its safety function following a design basis LOCA. The third main steam line valve should be classified seismic Category I and quality group classification in accordance with BTP's ASB 3-2 and ASB 3-3.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The main steam isolation valve leakage control system (MSIVLCS) includes [the source of the sealing medium, (if used)] pumps, valves, and piping to the points of connection or interface with the main steam lines. Based on the review of the applicant's proposed design criteria, design bases, safety classification of system

and components, and the requirements for operation of the system during loss-of-coolant accident conditions, the staff concludes that the design of the main steam isolation valve leakage control system is in conformance with the Commission's regulations as set forth in General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," General Design Criterion 4, "Environmental and Missile Design Bases," General Design Criterion 54, "Piping Systems Penetrating Containment," and meets the guidelines in Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.29, "Seismic Design Classification," Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Regulatory Guide 1.117, "Tornado Design Classification," Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment " and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," Branch Technical Position ASB 3-2, "Classification of Main Steam Components Other Than Reactor Coolant Pressure Boundary," and Branch Technical Position ASB 3-3, "Classification of BWR-6 Main Steam and Feedwater Components Other Than Reactor Coolant Pressure Boundary," and therefore is acceptable.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

4. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste -Containing Components of Nuclear Power Plants."

5. Regulatory Guide 1.29, "Seismic Design Classification."

6. Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants."

7. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

8. Regulatory Guide 1.117, "Tornado Design Classification."

9. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1, and MEB 3-1,

"Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

10. Branch Technical Position ASB 3-2, "Classification of Main Steam Components Other Than Reactor Coolant Pressure Boundary for BWR Plants."

11. Branch Technical Position ASB 3-3, "Classification of BWR/6 Main Steam and Feedwater Components Other Than Reactor Coolant Pressure Boundary."

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SECTION 7.1 INSTRUMENTATION AND CONTROLS - INTRODUCTION

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Auxiliary Systems Branch (ASB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The ICSB reviews Section 7.1 of the applicant's safety analysis report (SAR) which tabulates all safety-related instrumentation and control systems, their design bases, and the applicable acceptance criteria required for the safety of the plant during all modes of operation including transients and accidents. ICSB reviews this information as detailed in subsection III.

The secondary review branches (ASB, PSB, CSB, RSB) review the tabulations of safety-related systems for completeness, i.e., to verify that all safety-related systems within their respective areas of primary review responsibility have been identified. If systems other than those identified are deemed to be safety-related, this information is transmitted to ICSB.

This SRP section also includes evaluation of the proposed technical specifications given in SAR Chapter 16 to assure that they are adequate with regard to safety system settings, limiting conditions for operation, and periodic surveillance testing of instrumentation and controls.

II. ACCEPTANCE CRITERIA

The identification of safety-related systems is acceptable when it can be concluded that the integrated response of these systems assures the safety of the plant in normal operation, anticipated operational transients, and postulated accidents.

Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems," lists the criteria currently applicable to safety-related instrumentation and control systems

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(acceptance criteria for safety-related electric power systems are listed in Table 8-1). Conformance to these criteria does not necessarily establish the adequacy of the functional performance and reliability of these systems. However, omission of any of the criteria will in most cases be an indication of system inadequacy. Therefore, the identification of the criteria applicable to safety-related instrumentation and control systems is acceptable if it includes all of the criteria listed in Table 7-1, and if the SAR contains a statement to the effect that these criteria are implemented, at the operating license (OL) stage, or will be implemented, at the construction permit (CP) stage, in the design of these systems.

The fundamental bases for acceptance of the proposed technical specifications are that the limiting conditions for operation are such that sufficient equipment is required to be available for operation to meet the single failure criterion; that equipment outages, permissible for a short period of time, still leave available sufficient equipment to provide the protective function assuming no failures; and that the provisions of the technical specifications are compatible with the safety analyses.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

Safety-related systems fall into three categories: basic safety systems, auxiliary supporting systems, and other systems important to safety.

Basic safety systems are those that directly perform a protective function. Examples are the reactor trip system, the emergency core cooling system, the containment isolation system, and the containment spray system. The reactor trip system provides reactor protection by fast insertion of negative reactivity (control rods) when plant conditions approach design safety limits. All the other systems listed are engineered safety features (ESF) systems; their function is to mitigate the consequences of postulated design basis accidents.

Auxiliary supporting systems are those that must function to enable operation of the basic safety systems. Component cooling systems, service water systems, ventilation systems, and electric power systems which serve ESF and reactor trip components are examples of auxiliary supporting systems. These systems must meet the same criteria as the basic safety systems they support.

Other systems important to safety are those systems which operate to reduce the probability of occurrence of specific accidents, or to maintain the plant (including other safety systems) within the envelope of operating conditions postulated in the accident analyses as being required to assure full protection capability. Examples of this type of system are the cold loop startup control (interlocks) system, the accumulator tank

isolation valve control (interlocks, position indication, alarms) system, and the plant status and alarm systems that provide the operator with the information necessary for initiating manual protective action. These systems are primarily instrumentation and control systems characterized by having a functional interface with the operator. The same safety criteria apply. However, in application to these types of systems, the criteria are usually further defined by regulatory guides and in branch technical positions of the ICSB.

The ICSB review encompasses all of the instrumentation and control systems associated with all three categories of safety-related systems described above, with particular emphasis on the elements which constitute the protection system (as defined in IEEE Std 279) and the Class IE electric systems (as defined in IEEE Std 308). The safety-related electric power systems are covered in the SRP sections for Chapter 8 of the SAR. The SRP sections for SAR Chapter 7 are concerned only with the safety-related instrumentation and control systems.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The review of SAR Section 7.1 and applicable portions of the plant technical specifications is performed as follows:

1. ICSB will establish that all safety-related systems are identified, and that this identification does not conflict with the more detailed information provided in other sections of the SAR, particularly in Chapters 6 and 8 and in subsequent sections of Chapter 7. The definitions of safety-related systems presented above should be used as an aid in assessing the completeness of the identification. The secondary review branches (ASB, PSB, CSB, RSB) will confirm the identification of all safety-related systems within their respective areas of primary review responsibility. If systems other than those identified are deemed to be safety-related, this information should be transmitted to ICSB. Particular care should be exercised to assure that all systems postulated in the accident analyses (Chapter 15) as being required for safety are identified as safety-related systems.

2. ICSB verifies that other systems described in the SAR (particularly in Chapters 5, 6, 8, 9, 10, 11, and 15) but not identified by the applicant in Section 7.1 are not required for safety. The reviewer should obtain concurrence from the secondary review branches with regard to systems considered to be safety-related by ICSB, but which have not been identified as such by the applicant. Written requests for evaluation should be made to the secondary review branches when there are novel designs or significant differences of opinion.

3. ICSB verifies that the safety-related systems are categorized by supplier, i.e., those designed and supplied by the nuclear steam system supplier and those designed or supplied by others.

4. ICSB verifies that systems identical to those of reference plants that have recently received construction permits or operating licenses and those that differ from the reference plants are so identified.

5. ICSB verifies that for those systems that are different from those of the reference plants, the differences are described and justified to the extent necessary for an evaluation of their safety significance.

6. ICSB confirms that the criteria identified as being applicable to the design of safety-related instrumentation and control systems include those criteria listed in Table 7-1. This identification meets the applicable requirements of General Design Criterion 1, "Quality Standards and Records," of Appendix A of 10 CFR Part 50. General Design Criterion 1 also requires that, "Structures, systems and components important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed." Therefore, the SAR should include (1) a discussion regarding the applicability of each criterion listed, and (2) a statement to the effect that the criteria are implemented (OL) or will be implemented (CP) in the design of safety-related instrumentation and control systems.

7. ICSB verifies that technical design bases are provided (reference to other sections of the SAR is acceptable) for all the various functions of the protection system.

8. Applicable portions of the proposed plant technical specifications (SAR Chapter 16) are reviewed by ICSB and the secondary review branches to:
   a. Confirm the suitability of the safety limits, limiting safety system settings, and the limiting conditions for operation.
   b. Verify that the frequency and scope of periodic surveillance requirements are adequate.

For a CP review, it is only necessary to confirm that the applicant has identified those variables, conditions, or other items which have been determined to be probable subjects of the technical specifications (see 10 CFR §50.34(a)(5)). The applicant's justification for the selection of those items is evaluated with special attention to any that may significantly influence the final design. The specific provisions of the proposed technical specifications are not approved during the CP review. However, any specific provisions which are known to be unacceptable or which may influence acceptance of the preliminary design of the plant should be brought to the applicant's attention and, if appropriate, included in that portion of the staff's safety evaluation report pertaining to the design of the affected systems.

For an operating license review, the proposed technical specifications are reviewed and evaluated in depth in accordance with the requirements of 10 CFR §50.36. For the ICSB areas of review, a check is made that the limiting conditions for operation (LCO) agree with the surveillance requirements, i.e., for each system or component that is the

subject of an LCO, there must be corresponding surveillance requirements. Each system or component that performs a function for which credit is taken in the accident analyses should be the subject of an LCO. The limiting safety system settings should be in accordance with the values assumed in the accident analyses, including appropriate allowances for instrument error, drift, etc. If the acceptance of the design of a particular system is based upon required plant conditions or particular operating procedures, such requirements should be included in the final technical specifications and, if appropriate, noted in that portion of the staff's safety evaluation report pertaining to the design of the affected system.

IV. EVALUATION FINDINGS

ICSB verifies that sufficient information has been provided and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The applicant has identified the safety-related instrumentation and control systems and the applicable safety criteria and has documented his intent to design and implement these systems in accordance with the criteria. It is concluded that implementation of these systems in accordance with the criteria provides assurance that the plant will perform as designed in normal operation, anticipated operational transients, and postulated accident conditions, and meets the applicable requirements of General Design Criterion 1."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."*

*All references for this plan are included in Standard Review Plan Table 7-1.

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TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROLS

Table 7-1 contains the acceptance criteria for the SRP sections of Chapter 7. These acceptance criteria include the applicable General Design Criteria, IEEE standards, Regulatory Guides, and Branch Technical Positions (BTP) of the Instrumentation and Control Systems Branch (ICSB). The applicability of these criteria to specific sections of Chapter 7 is indicated by an X in the matrix listing of criteria and SAR sections. The BTP listed in Table 7-1 are contained in Appendix 7-A to the Chapter 7 SRP section.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

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Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS - TABLE 7-1

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
1. 10 CFR Part 50									
a. 10 CFR §50.34	Contents of Application: Technical Information	X	X	X	X	X	X	X	
b. 10 CFR §50.36	Technical Specifications	X	X	X	X	X	X		
c. 10 CFR §50.55a	Codes and Standards	X	X	X	X	X	X	X	
2. General Design Criteria (GDC), Appendix A to 10 CFR Part 50									
a. GDC 1	Quality Standards and Records	X	X	X	X	X	X		
b. GDC 2	Design Bases for Protection Against Natural Phenomena	X	X	X	X	X	X		
c. GDC 3	Fire Protection	X	X	X	X	X	X		
d. GDC 4	Environmental and Missile Design Bases	X	X	X	X	X	X		
e. GDC 5	Sharing of Structures, Systems, and Components	X	X	X	X	X	X		
f. DGC 10	Reactor Design	X	X	X	X	X	X		
g. GDC 12	Suppression of Reactor Power Oscillations	X	X			X		X	
h. GDC 13	Instrumentation and Control	X	X	X	X	X	X	X	
i. GDC 15	Reactor Coolant System Design	X	X			X	X	X	
j. GDC 19	Control Room	X	X	X	X	X	X	X	
k. GDC 20	Protection System Functions	X	X	X	X	X	X		
l. GDC 21	Protection Systems Reliability and Testability	X	X	X	X	X	X		
m. GDC 22	Protection System Independence	X	X	X	X	X	X		
n. GDC 23	Protection System Failure Modes	X	X	X	X	X	X		
o. GDC 24	Separation of Protection and Control Systems	X	X	X	X	X	X	X	
p. GDC 25	Protection System Requirements for Reactivity Control Malfunctions	X	X			X			

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
q. GDC 26	Reactivity Control System Redundancy and Capability	X	X		X	X		X	
r. GDC 27	Combined Reactivity Control Systems Capability	X	X		X	X		X	
s. GDC 28	Reactivity Limits	X	X			X	X	X	7.6 Interlocks only
t. GDC 29	Protection Against Anticipated Operational Occurrences	X	X	X	X	X	X	X	
u. GDC 33	Reactor Coolant Makeup	X			X	X			
v. GDC 34	Residual Heat Removal	X		X	X	X	X		
w. GDC 35	Emergency Core Cooling	X	X	X		X	X		
x. GDC 37	Testing of Emergency Core Cooling System	X	X	X		X	X		
y. GDC 38	Containment Heat Removal	X		X		X	X		
z. GDC 40	Testing of Containment Heat Removal System	X		X		X	X		
aa. GDC 41	Containment Atmosphere Cleanup	X		X		X	X		
bb. GDC 43	Testing of Containment Atmosphere Cleanup Systems	X		X		X	X		
cc. GDC 44	Cooling Water	X		X		X	X		
dd. GDC 46	Testing of Cooling Water System	X		X		X	X		
ee. GDC 50	Containment Design Basis	X		X		X	X		
ff. GDC 54	Piping Systems Penetrating Containment	X		X		X	X		
gg. GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	X		X		X	X		
hh. GDC 56	Primary Containment Isolation	X		X		X	X		
ii. GDC 57	Closed Systems Isolation Valves	X		X		X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:									
a. IEEE Std. 279 (ANSI N42.7)	Criteria for Protection Systems for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See 10 CFR §50.55a(h) and Reg. Guide 1.62.
b. IEEE Std 308	Criteria for Class IE Electric Systems for Nuclear Power Generating Stations	X			X	X	X		See Reg. Guide 1.32.
c. IEEE Std 317	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.63. SRP Section 3.11.
d. IEEE Std. 336 (ANSI N45.2.4)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.30.
e. IEEE Std 338	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X		See Reg. Guide 1.118.
f. IEEE Std 344 (ANSI N41.7)	Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations	X	X	X	X	X	X		See Reg. Guide 1.100 SRP Section 3.10.
g. IEEE Std 379 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X	X	See Reg. Guide 1.53.
h. IEEE Std 384 (ANSI N41.14)	Criteria for Separation of Class IE Equipment and Circuits	X	X	X	X	X	X	X	See Reg. Guide 1.75.

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
4. Regulatory Guides (RG)									
a. RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	X			X	X	X		
b. RG 1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	X		X		X			
c. RG 1.11	Instrument Lines Penetrating Primary Reactor Containment	X	X	X	X	X	X		
d. RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	X	X		
e. RG 1.29	Seismic Design Classification	X	X	X	X	X	X		SRP Section 3.10
f. RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	X	X	X	
g. RG 1.32	Use of IEEE Std 308 "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"	X			X	X	X		
h. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	X	X		Use in conjunction with Position 3, RG 1.17.
i. RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X	X	X	X	X	X		
j. RG 1.62	Manual Initiation of Protection Actions	X	X	X	X		X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
k. RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plant	X	X	X	X	X	X	X	
l. RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X	X	X	X	X	X	
m. RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2	X	X	X	X	X	X	X	
n. RG 1.75	Physical Independence of Electric Systems	X	X	X		X			
o. RG 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	X					X		
p. RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X	X	X	X	X	X		SRP Section 3.11.
q. RG 1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	X		X					
r. RG 1.12	Instrumentation for Earthquakes	X					X		
s. RG 1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	X					X		
t. RG 1.67	Installation of Overpressure Protection Devices	X					X		
u. RG 1.80	Pre-operational Testing of Instrument Air	X		X	X		X		SRP Section 9.

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
v. RG 1.95	Protection of Nuclear Power Plant Control Room Operators Against Accidental Chlorine Releases	X					X		
w. RG 1.97	Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and following an Accident	X				X			
x. RG 1.100	Seismic Qualification of Electrical Equipment for Nuclear Power Plants	X	X	X	X	X	X		SRP Section 3.10.
y. RG 1.105	Instrument Spans and Setpoints	X	X	X	X	X	X		
z. RG 1.118	Periodic Testing of Electric Power and Protection Systems	X	X	X	X	X	X		
aa. RG 1.120	Fire Protection Guidelines for Nuclear Power Plants	X	X	X	X	X	X	X	SRP Section 3.10.
5. Branch Technical Positions (BTP) ICSB									
a. BTP ICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X	X	X	X		X		DOR Responsibility.
b. BTP ICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System	X			X		X		
c. BTP ICSB 4 (PSB)	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	X			X		X		
d. BTP ICSB 5	Scram Breaker Test Requirements - Technical Specifications	X	X						
e. BTP ICSB 9	Definition and Use of "Channel-Calibration" - Technical Specifications	X	X		X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
f. BTP ICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X	X		X	X	X		Replaced by Reg. Guide 1.100
g. BTP ICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	X	X	X					
h. BTP ICSB 13	Design Criteria for Auxiliary Feedwater Systems	X		X					
i. BTP ICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	X	X					X	
j. BTP ICSB 15 (PSB)	Reactor Coolant Pump Breaker Qualification	X	X						
k. BTP ICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors	X	X						
l. BTP ICSB 18 (PSB)	Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves	X		X	X		X		
m. BTP ICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems	X		X			X		
n. BTP ICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	X		X	X		X		
o. BTP ICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X	X	X		
p. BTP ICSB 22	Guidance for Application of Req. Guide .122	X	X	X	X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
q. BTP ICSB 23	Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown	X				X			Replaced by Reg. Guide 1.97.
r. BTP ICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times	X	X	X	X		X		Replaced by Reg. Guide 1.118.
s. BTP ICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole	X		X	X				
t. BTP ICSB 26	Requirements for Reactor Protection System Anticipatory Trips	X	X						
u. BTP ICSB 27	Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves	X		X	X		X		Replaced by Reg. Guide 1.106

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SECTION 7.2 REACTOR TRIP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Auxiliary Systems Branch (ASB)
Core Performance Branch (CPB)
Mechanical Engineering Branch (MEB)
Power Systems Branch (PSB)
Quality Assurance Branch (QAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

ICSB reviews Section 7.2 of the applicant's safety analysis report (SAR), which describes the reactor trip system (RTS). The reactor trip system, which is part of the reactor protection system, includes those power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices, that are required to initiate reactor shutdown. The RTS is designed to initiate automatically the reactivity control system (control rods), to assure that specified acceptable fuel design limits are not exceeded. It also includes those safety-related portions of control systems, the actions of which inhibit or limit the response of the reactivity control system to ensure that fuel design limits and safety limits are not exceeded.

Although the design configurations of RTSs for nuclear reactors vary significantly, it is possible by use of the diagram in Figure 7.2-1 to define the RTS of each nuclear steam supply system (NSSS) to the extent necessary for the purpose of identifying the ICSB primary review responsibility.

As shown in Figure 7.2-1, the RTS includes several sensors (usually four) to measure each parameter such as neutron flux, primary system pressure, reactor outlet temperature, etc. These parameters are detected by sensors of various principles and types that provide electrical signals, mostly at low current or voltage levels. The sensors are located at many locations throughout the plant. It is necessary to determine that each location is suitable for the type of sensor used and that its transmission circuitry (channel) is properly routed to the RTS cabinets in which the electronic signal conditioning equipment is located. Most often, sensors are mounted on local racks and panels. Their arrangement should be considered in the review. For example, consideration should be given to the routing of sensing lines from the process system taps to the sensors, the sensor mountings on racks, and the arrangement of local racks and panels within the

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plant. The paths of transmission circuitry include routing through containment electrical penetrations and into the cable spreading room. These regions deserve special review attention with regard to ascertaining RTS compliance with the acceptance criteria of subsection II.

The reactor trip system cabinets that include signal conditioning equipment, logic arrangements, test circuitry, indicators, alarms, and other features are the focal point of the RTS. The cabinets are usually located in the control room area. In addition to reviewing the cabinets and their contents against the acceptance criteria, the reviewer must show that the cabinets are not vulnerable to significant degradation from external influences. Other significant RTS cabinets include those that contain the system trip actuation devices themselves. The actuation devices and the power circuitry to the actuated devices (control rod drives) are also within the scope of the ICSB review; however, the control rod poison sections, the power sources for the actuation devices, and the control rod drives are reviewed by others.

The power supply for the RTS is included in the ICSB review to the extent that the review must show that loss of power would not result in RTS failure to function. The review need not address the capability of the power supplies, usually motor-generator sets, to supply power. However, uniqueness of voltage and frequency requirements for certain RTS motor-generator sets and power supplies must be considered by PSB.

Testability of the RTS must be reviewed to ensure that the entire system is fully testable. The ICSB reviewer must ascertain that the test circuitry and test methods used do not compromise the independence of redundant circuits and equipment and do, in fact, enhance RTS reliability. This concern is particularly significant for newer solid state designs incorporating automatic test features.

Another review area of significance includes the interlock circuits that are provided to inhibit control rod motion. These are actuated from safety-related control system sensors such as those that monitor control rod position, turbine trip, etc. Also, protective interlocks actuated from loop isolation valve switches that are used to reset RTS parameter trip levels to more conservative values must be reviewed, along with manual selector switches that are also used to reset protection system trip levels as required for other modes of operation than the normal full power operating mode. These are shown schematically in Figure 7.2-1.

The descriptive information, electrical schematics (for the operating license stage only), logic diagrams, and physical arrangement drawings are reviewed. The objectives are to determine, on the basis of the most recent diagrams available, that the RTS satisfies the acceptance criteria and to determine that the RTS will perform its intended function during accident conditions and other transient conditions identified in the safety analysis report (SAR) accident analyses. This capability must be maintained during all plant operating modes including startup power operation, shutdown, and refueling, as defined by the technical specifications.

The depth of review for a plant at the construction permit (CP) stage is limited. For a construction permit, design criteria and preliminary designs are reviewed in order to establish a basis for acceptance. The level of detail need only be sufficient to provide reasonable assurance that the final design will conform to the design bases and that the design bases themselves provide an adequate margin for plant safety. For an operating license (OL), the final design diagrams and results of analyses are reviewed to determine that the required safety functions can be accomplished.

The review also includes evaluation of the proposed technical specifications to assure their adequacy. Refer to SRP Section 7.1 for the considerations involved.

In summary, the primary review area within the scope of the ICSB for SAR Section 7.2 includes:

1. The descriptive information, design bases, and analyses for the reactor trip system.

2. The descriptive information and design bases for supporting systems interfacing with and essential to the operation of the reactor trip system.

In cases where the design is similar to that of plants previously reviewed, the reviewer may determine that it is not necessary to review every facet of the design but may instead select and place emphasis on the most critical areas. Conversely, when concepts that have not previously been reviewed by the staff are received for review, evaluation considerations beyond those outlined in this section may be applied as necessary to assure that the proposed designs will function properly and meet all applicable requirements.

To assure that the auxiliary supporting systems that are essential to RTS operation will adequately maintain the required environmental conditions in areas of the plant where RTS equipment is located, ASB support is required in the evaluation of cooling systems, heating and air conditioning systems, etc. The ASB provides assistance in determining that the RTS will be capable of performing its function with auxiliary supporting systems degraded to their limiting conditions for operation. The auxiliary systems are described in SAR Chapters 9 and 10, for which ASB has primary review responsibility.

Assistance is required from the CPB in reviewing the reactivity control aspects of the RTS, including negative reactivity available in control rods, allowable reactivity insertion or withdrawal rates, and reactivity distributions throughout plant life. The CPB reviews the placement of neutron sensors with regard to measurement of the flux spatial dependence, the flux magnitude, and calibration effects for all operating modes throughout core life. CPB assistance is also required to establish technical specifications for core protection instrumentation with regard to limiting conditions for operation and limiting safety system settings. The plant nuclear design is discussed in SAR Section 4.3, for which CPB has primary review responsibility.

ICSB requires support from the MEB to review seismic qualification tests and supporting analysis for RTS equipment. The MEB review responsibilities in this regard are discussed in SRP Section 3.10.

The RTS design and construction must be carried out in accordance with the quality assurance requirements of 10 CFR Part 50, General Design Criterion 1 and Appendix B. QAB assistance is required to make this determination. QAB also determines that the quality assurance program documentation required of applicants and the proposed QA/QC organizations are acceptable. The QAB review responsibilities are discussed in SRP Sections 17.1 and 17.2.

To assure that the location, number, and ranges of sensors provided to monitor the performance of the reactor heat transfer systems and related equipment are adequate, the ICSB requires RSB support. RSB assistance is also required to establish technical specification requirements for heat transfer system instrumentation with regard to limiting conditions for operation and limiting safety system settings. The RSB primary review responsibilities are discussed in SRP Sections 4.4, 6.3 and 15.0.

I. ACCEPTANCE CRITERIA

In general, the reactor trip system is acceptable if it includes adequate redundancy; meets the single failure criterion; has the capacity and capability to safely and reliably shut down the reactor; is fully testable; is capable of functioning during and after design basis events and accidents; and satisfies applicable requirements of the regulations and the recommendations of Institute of Electrical and Electronic Engineers (IEEE) standards, regulatory guides, and branch technical positions. Subsection V lists those regulations, standards, guides, and positions used by the reviewer as aids in ascertaining that the above criteria have been met. Subsection III discusses the application of these evaluation guides to the review.

The general design criteria and IEEE Std 279 set forth requirements that must be met by all RTS designs. Supporting auxiliary systems must also satisfy these requirements. Appendix A to this SRP section provides the reviewer with a summary of the use of IEEE Std 279 in the review.

The regulatory guides and branch technical positions set forth acceptable methods of implementing criteria and are not requirements. They serve to resolve problems by proposing particular solutions. Industry standards and topical reports referenced in a SAR may be used as a basis for approval of a design. However, acceptability of the standards and topical reports referenced, but not previously reviewed, must be determined in order to complete the review of the SAR.

Acceptance criteria for specific areas of RTS design are as follows (a complete listing of these criteria is included in Table 7-1, attached to Chapter 7.0 of this Standard Review Plan):

1. System Redundancy Requirements

General Design Criteria 20 through 29 set forth requirements with regard to functional redundancy considerations. General Design Criteria 2, 3 and 4 set forth the external considerations that must be reviewed to assure that redundancy is not compromised. IEEE Std 279 and IEEE Std 379 are also useful to the reviewer in determining redundancy requirements for the RTS.

2. System Conformance with the Single Failure Criterion

The General Design Criteria applicable to the preceding discussion on system redundancy requirements (II.1, above) apply equally to system conformance to the single failure criterion. In addition to the general requirements of these regulations, Regulatory Guide 1.53 (as it relates to IEEE Std 379) and IEEE Std 279, paragraphs 4.2, 4.7.3, 4.7.4, 4.7.4.1, 4.7.4.2 and 4.11, explicitly address the single failure criterion and form the basis for judging system conformance to the single failure criterion. Also, see Appendix A to this SRP section for additional guidance.

3. System Capability and Reliability

The general requirements for RTS capability are included in General Design Criteria 20 through 29. With the exception of RTS response time, the analyses performed by the CPB and described in SRP Section 4.3 serve as the basic acceptance criteria for capability. The basis for system response time acceptance is established in the SAR, usually in Chapters 7 and 15. RTS reliability considerations and their conformance to General Design Criterion 21 are based on analyses, as documented in NSSS topical reports, and on testing and operating experience with given hardware.

4. System Testability

The criteria used to judge system testability and conformance with General Design Criterion 21 are basically those contained in IEEE Std 279, IEEE Std 338 (as endorsed in Regulatory Guide 1.118), and Regulatory Guide 1.22. In addition, initial qualification of the system is evaluated on the basis of IEEE Std 336, IEEE Std 344 (as endorsed in Regulatory Guide 1.100) and Regulatory Guide 1.68. Also, an acceptable design must satisfy the concerns of Regulatory Guide 1.47 and Branch Technical Position ICSB 21.

5. System Capability During and Following Design Basis Events

The method used to assure that the RTS will be capable of performing its protective function during and following design basis accidents is that of equipment qualification for the conditions postulated to accompany the events.

General Design Criteria 2, 3 and 4 identify events of concern and state acceptance objectives. IEEE Std 344 (as modified by Regulatory Guide 1.29 and Regulatory Guide 1.100) for seismic qualification, IEEE Std 317, IEEE Std 323, and IEEE Std 336 for environmental qualification provide the acceptance criteria. IEEE Std 336 is augmented by Regulatory Guide 1.30, IEEE Std 317 is augmented by Regulatory Guide 1.63, and IEEE Std 323 is augmented by Regulatory Guide 1.89.

6. Identification of Control Panels, Racks, Equipment, Cables, and Cable Trays

The method used for identifying RTS cables and cable trays as safety-related equipment in the plant, and the identification scheme used to distinguish between redundant equipment, racks, panels, cables, and cable trays, are evaluated on the basis of Section 5.1.2 of IEEE Std 384, as endorsed in Regulatory Guide 1.75. IEEE Std 279, paragraph 4.22, also addresses identification criteria.

7. Separation of Equipment, Cables, and Cable Trays

IEEE Std 384, as endorsed in Regulatory Guide 1.75, provides a basis for review and acceptance of the separation criteria presented in the SAR.

8. Vital Supporting Systems

The auxiliary systems that are required to assure RTS functional capability should satisfy the same acceptance criteria as the RTS.

9. Technical Specifications

The acceptance criteria for technical specifications are identified in 10 CFR §50.34 and 50.36. Usually the most recently licensed plant of the type being reviewed serves as a model for the technical specifications. Standard technical specifications are also in preparation at this time. Refer to SRP Section 7.1 for technical specification considerations.

For those areas of review identified in subsection I as being the responsibility of other branches, the acceptance criteria and their application are included in the appropriate SRP sections for which these branches have primary responsibility. These are criteria that are used by both primary and secondary review branches as the basis for accepting a design. As they relate to the RTS, some of these criteria and their application are presented below.

In assuring the adequacy of the seismic design of Category I instrumentation and electrical equipment, both the MEB and ICSB perform reviews to ascertain that the proposed design satisfies IEEE Std 344 as supplemented by Regulatory Guide 1.100.

To assure that the requirements of General Design Criterion 1 and Appendix B of 10 CFR Part 50 are met in the reactor trip system, the quality assurance program for the RTS Class 1E instrumentation and electrical equipment must satisfy the concerns of IEEE Std 336, as endorsed in Regulatory Guide 1.30.

III. REVIEW PROCEDURES

The main objectives in the review of the reactor trip system are to determine that this system includes the required redundancy, satisfies electrical and physical independence requirements and the single failure criterion, has the capability and reliability required, is testable, is capable of performing its function during and following design basis events, and can safely shut down the reactor in conformance with all the general design criteria requirements for RTS and the requirements documented in the accident analysis chapter of the safety analysis report.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

In the construction permit (CP) review, the descriptive information, including system safety design bases and their relationship to the acceptance criteria, preliminary analyses, electrical single line diagrams, preliminary physical arrangement drawings, functional logic diagrams, and functional piping and instrument diagrams (P&IDs,) are examined to determine that there is reasonable assurance that the final design will meet the above objectives. Included in this review, the design criteria for establishing trip setpoints must be evaluated to show conformance to the following guidelines:

1. The range selection for instrumentation shall be such as to exceed the expected range of the process variable being monitored.

2. The accuracy of all the safety trip points will not be numerically larger than the accuracy that was assumed in the accident analysis.

3. The trip setpoints should be located in that portion of the instrument's range which is most accurate and must be located in a region with the required accuracy.

4. All safety trip points will be chosen to allow for the normal expected instrument system setpoint drift such that the technical specification limit will not be exceeded.

5. Verification of the above criteria shall be demonstrated as a part of the qualification test program required by IEEE Std 323.

At the operating license (OL) stage of review, these objectives are verified in the review of final electrical schematics and physical arrangement drawings. In addition, a site visit is conducted to assure that the design objectives have, in fact, been implemented in accordance with the design bases and criteria. Appendix 7-B to SRP Chapter 7.0 contains a typical site visit agenda.

This section describes the method and reasoning to be employed by the reviewer in making a determination as to RTS acceptability. For the purpose of illustration, the RTS system as presented in Figure 7.2-1 is shown as being comprised as two identical, redundant subsystems.

Prior to reviewing Section 7.2 of the SAR, the following background information should be briefly reviewed, in addition to the balance of Chapter 7:

Chapter 1 of the SAR, to become familiar with the general operation of the plant, from both the safety and the operational standpoints.

Chapter 4, the reactor design, in particular the nuclear design, Section 4.3, and the thermal and hydraulic design, Section 4.4.

Chapter 5, on the design of the reactor coolant system, Sections 5.1, 5.2.1 and 5.2.2.

Chapter 6, to note the engineered safety features provisions.

Chapter 15, to become familiar with the representative types of events for which analyses have been documented. In particular, the effects of failures of the protective functions, and the assumptions and initial conditions that form the bases of the accident analyses are noted.

Chapter 16, to become familiar with limiting conditions for operation, limiting safety system settings (i.e., trip setpoints), and surveillance requirements that pertain to the RTS.

Chapter 17, to note the quality assurance considerations addressed.

The single most relevant document used in the review of the RTS is IEEE Std 279. Conformance of the RTS to the design requirements stated in Sections 3 and 4 of this standard, together with conformance to the requirements of the general design criteria and the functional requirements derived from the accident analyses, will result in an acceptable design. Guidance on the use of IEEE Std 279 is provided in Appendix A to this SRP section. The general methodology by which the reviewer conducts his review is outlined below by addressing "key concerns" such as redundancy, independence, single failures, capability, and testing.

1. System Redundancy Requirements

With the assistance of the CPB and the RSB, as needed, ICSB determines that the system redundancy requirements are satisfied. Generally, a minimum degree of redundancy of one satisfies RTS requirements. Most RTS parameters are monitored by four sensor channels and only two of four channels are required to initiate the RTS logic channel protective action.

Where it is determined that the spatial dependence of a parameter requires several sensor channels to assure core protection, the redundancy requirements are determined for the individual case. Once design adequacy is established, the reviewer must relate the design requirement to the limiting conditions for operation in the technical specifications. In certain designs, for example, adequate monitoring of core power requires a minimum number of sensors arranged in a given configuration to permit unrestricted power operation. When, because of system degradation, the

minimum number of sensors are not available, operation must be restricted. This aspect of redundancy must be dealt with in coordination with the CPB to establish conditions of restricted operation.

Another area where the redundancy requirement of the RTS may have to be defined on an individual core basis is the allowable power operation for reactor coolant systems that have loop isolation valves. Here, the redundancy of the instrumentation provided on the reactor coolant system piping, and on steam generators in the case of pressurized water reactors (PWRs), must be reviewed with the RSB to determine whether the reactor system instrumentation redundancy (not channel redundancy) requirement has been degraded below that on which the accident analyses are based if isolation valves are closed.

With regard to redundancy requirements considered strictly from an electrical point of view, it is only necessary to assure that at least two redundant logic trains (minimum degree of redundancy of one) are provided to initiate reactor trip. From this standpoint the review may be reduced to a simple analysis in which redundant paths from sensors to logic and to actuation devices are identified to assure that the RTS functional requirements are met. It is pointed out that redundancy may be accomplished by equipment that is diverse in principle so long as the same level of protection is provided.

In this discussion on RTS redundancy, it is appropriate to reference Figure 7.2-1. Notice that for required protective functions, the RTS sensors, initiation devices, logic matrices, and actuation and actuated devices all must be redundant. Also note that modules of one channel must not affect those of another channel.

Another area that must be reviewed with regard to redundancy has to do with the measures to be included in nuclear power plants to deal with ATWS events. These measures must be reviewed to assure that they are unaffected by failures that could disable the RTS.

2. System Conformance with the Single Failure Criterion

In evaluating the adequacy of the RTS system in meeting the single failure criterion, both electrical and physical independence must be considered.

a. Electrical Independence

To assure electrical independence, the design bases governing the electrical independence of redundant sensors, logic elements, and actuation channels are required to satisfy not only paragraph 4.6 of IEEE Std 279, which states that, "channels that provide signals for the same protection function shall be independent, and the likelihood of interaction between channels is considered," but also the requirement of paragraph 4.7.2. This paragraph requires that, "the transmission of signals from protection system channels that are used for other purposes (non-protective), such as control or readout and indication,

are properly isolated to ensure that no credible failure at the output of an isolation device shall prevent the associated protective channel from meeting performance requirements." Examples of credible failures at the output of isolation devices are provided in paragraph 4.7.2.

b. Physical Independence

To assure physical independence, the design bases governing the physical separation of redundant equipment including sensors, cables, cable trays, racks, panels, and control boards are evaluated in accordance with Regulatory Guide 1.75, "Physical Independence of Electric Systems." This regulatory guide sets forth acceptance criteria for the physical separation of circuits and electrical equipment that is included in the RTS.

Another review objective is to determine whether the RTS is located in seismic Category I structures. In certain designs, RTS sensors may be located in non-seismic Category I structures such as the turbine building. For these special cases, the reviewer must assure that the most reasonable installation of sensors and circuits is provided in regard to physical protection against damage from a seismic event. Further guidance is provided by Branch Technical Position ICSB 15.

c. Single Failure Criterion

To assess the RTS acceptability with regard to the single failure criterion, IEEE Std 379 and Regulatory Guide 1.53 are used. Again, as was the case for redundancy requirements, review for compliance with the single failure criterion may be reduced to an analysis in which it is determined that the system can perform all protective functions concurrent with failure of any sensor, logic circuitry and components that meet the single failure criterion. IEEE Std 279, paragraph 4.2, provides an additional example of single failure criterion application.

With regard to power requirements, the RTS must be reviewed to assure that no failure of a power supply will result in maintaining power to the system such that the protective function (trip) of the RTS is negated (fail-safe design). For example, loss of power to a sensor channel should cause a channel trip. Similarly, a loss of power to a logic element or actuator channel should result in a trip. Exception to this latter rule may be taken, as long as the single failure criterion is satisfied and the power sources required are designed as Class 1E power systems.

The RTS logic matrices should be reviewed to determine whether redundant circuitry includes the contacts of relays or switches in mutually redundant logic ciruits. This task can be accomplished during the OL detailed drawing review. When violations of the single failure criterion are found, they are to be identified to the applicant and corrected. The staff safety evaluation

report should discuss the final disposition of designs that are revised to satisfy the acceptance criterion.

The RTS equipment arrangement must be reviewed to assure that no single credible event will result in a loss of redundant circuits or equipment. This matter is discussed further in III.5.b, below.

3. Identification of Control Boards, Equipment, Cables, and Cable Trays

To determine that the identification scheme used for Class 1E equipment, cables, and raceways in the plant and Class 1E internal wiring in the control boards is consistent with Regulatory Guide 1.75, the criteria proposed for identifying Class 1E wiring cables and cable trays are reviewed. This includes such criteria as those for distinguishing between safety-related cable trays of different channels, non-Class 1E cable which is run through Class 1E cable trays, and non-Class 1E cable which is not physically associated with any Class 1E division. IEEE Std 279, paragraph 4.22, also discusses identification. Color coding is a preferred method of identification. In multi-unit paths that share source spaces, it is particularly important to retain unit identifications along with channel identification.

4. System Testing and Inoperability Surveillance

The proposed preoperational and initial startup test programs for the RTS and its supporting systems are reviewed to verify that the proposed programs are consistent with the requirements set forth in IEEE Std 279, IEEE Std 308 (as augmented by Regulatory Guide 1.32), and Regulatory Guides 1.22, 1.68 and 1.118.

The descriptive information as supplemented by functional logic diagrams (CP and OL) and electrical schematics (OL) are reviewed to verify that the design has the necessary provisions to permit testing of the RTS on a periodic basis when the reactor is in operation. The reviewer is guided by the recommendations set forth in Regulatory Guide 1.22 and IEEE Std 279, paragraph 4.10, in arriving at an acceptable method of periodic testing of actuation devices (e.g., solenoids, breakers) and actuated equipment (control rods). The same guidance is used to review testability of all modules, relays, permissives, bypasses, and safety-related control devices.

The descriptive information (CP and OL) and the design implementation as depicted on electrical drawings (OL) of the means proposed for automatically indicating, at the system level, bypassed or deliberately inoperable RTS protection channels are reviewed to ascertain that the design is consistent with Regulatory Guide 1.47 as supplemented by Branch Technical Position ICSB 21 and with IEEE Std 279, paragraph 4.13.

5. Other Matters

a. The Technical Specification considerations for the RTS are outlined in SRP Section 7.1

b. The ASB reviews supporting systems such as heating and ventilating component cooling water, service water, etc., to assure that failure of these supporting systems will not result in loss of RTS function as result of a degraded environment. It is necessary to assure that those systems required to maintain environmental conditions within the envelope for which the RTS equipment and circuits were designed and qualified be monitored for performance. Examples of such systems include control room and switchgear room heating and ventilating systems.

THE ASB should also assist in determining hazardous conditions that might follow failure of nonsafety equipment in regions where RTS components and circuits are located. Specific failures must include, as a minimum, the following: fire, missiles, flooding, jet impingment from pipe breaks, and damage that may be caused by failure of non-seismic Category I structures and components. The ICSB relates these conditions to the ability of the RTS to retain functional capability.

c. To assure that the RTS provides adequate core protection, the CPB should confirm that the accident analyses of SAR Chapter 15 have addressed the requirements of IEEE Std 279, Section 3, "Design Basis." To accomplish this task, it is necessary to confirm that the accident analyses have taken into consideration such matters as spatial dependences, operational limits and margins, transient ranges, system response times, and signal and instrument accuracies.

d. The MEB has primary responsibility for assuring that the seismic design of Category I instrumentation and electrical equipment satisfies appropriate requirements. These include IEEE Std 344 and Regulatory Guide 1.100. ICSB supplements the MEB by reviewing the description of the seismic qualification test program (CP) and the results of such tests and analyses (OL) that demonstrate the capability of Class 1E instrumentation, control devices, and associated circuits to withstand the effects of seismic event. An integrated review is required.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The reactor trip system includes the initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices utilized to implement reactor shutdown. The scope of the review included the descriptive information (CP and OL), functional logic diagrams (CP and OL), functional instrumentation and electrical diagrams (CP and OL), and preliminary (CP) and final (OL) physical arrangement drawings and schematics. The review has included the applicant's design bases and their relation to the proposed design for the reactor trip system. The review has

also included the proposed means for identification of cables and equipment, periodic testing capability, and the qualification test program (CP) and the results (OL) for demonstrating the suitability of the reactor trip system.

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the reactor trip system and vital supporting systems to the Commission's regulations as set forth in the General Design Criteria and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

"On the basis of our review we have concluded that the reactor trip system conforms to applicable regulations, guides, technical positions, and industry standards, and is acceptable."

REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (ICSB)."

3. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

4. Regulatory Staff, "Technical Report on Anticipated Transients Without Scram," WASH-1270, U.S. Atomic Energy Commission, September 1973.

APPENDIX A

STANDARD REVIEW PLAN SECTION 7.2

USE OF IEEE STD 279 IN THE REVIEW OF THE RTS

This appendix discusses the requirements of IEEE Std 279, Section 4, as they are used in the review of the RTS.

1. Section 4.1 - This section requires that the RTS perform automatically and with precision and reliability. These requirements must be met over the full range of transient and steady-state conditions. The system must meet these requirements in any environmental condition expected during plant operation in which the applicant's accident analyses take credit for the function performed by the RTS. Other criteria which set forth similar requirements are General Design Criteria 2, 4, 10, 13, 20, 21 and 29.

   a. Automatic initiation is required for all protective functions. Manual initiation is also provided and is a requirement. (See Section 4.17 and Regulatory Guide 1.62.)

   b. The precision required in the RTS is that assumed in the accident analyses. (Precision requirements are identified in Section 3.9 of IEEE Std 279.)

   c. Quantitative reliability information for RTS is often presented in NSSS topical reports. The reliability requirements for RTS are primarily satisfied by related reactor operating experience. The staff is actively pursuing the question of RTS reliability and the potential for RTS loss of function. (See Reference 5, Section V of SRP Section 7.2.)

   d. The requirements for precise and reliable operation suggest that the RTS design should avoid unnecessary complexity. "Unnecessary complexity" is a difficult judgment: the reviewer should discuss his concerns with the system designer in detail, and should consult with the section leader and branch chief on this matter.

2. Section 4.2 - This section requires that the reviewer examine several different aspects of each single failure to determine its effect.

   a. The first step in a single failure analysis is to identify components that are not seismic Category I, those that are not qualified for accident environments, and those that serve both safety and nonsafety systems. Each of the nonqualified and nonsafety-grade components and systems are assumed to fail to function if failure adversely affects RTS performance, and are assumed to function if functioning adversely affects RTS performance.

b. The consequences of events for which the RTS is designed to provide a protective function are examined. All failures in the RTS that can be predicted to occur as a result of the events are assumed to occur if such events adversely affect RTS performance. In general, the lack of equipment qualification may serve as a basis to assume failures.

c. After assuming the failures of nonsafety-grade, nonqualified equipment and those failures in the RTS caused by an event, any other single failure is arbitrarily assumed and the resultant performance of the RTS is analyzed to assure that the minimum protective function will be performed.

d. The single failure criterion applies to all electric equipment. No distinction is made between active and passive components.

e. IEEE Std 379 and Regulatory Guide 1.53 are used for additional insight to single failure criterion analysis.

3. Section 4.3 - There are no specific criteria to judge the quality of the equipment used in the RTS. However, Appendix B to 10 CFR Part 50 provides some guidance from which a judgment may be made of the quality of equipment required for the RTS.

4. Section 4.4 - It is verified that each component and module has been qualified for normal, upset (i.e., operational transient), and accident environments at its installed location. This applies to all normal and upset conditions, but only to those accident conditions where the components and modules provide a protective function. The components must provide the accuracy, range, and response times required by the accident analyses. SRP Sections 3.10 and 3.11 discuss equipment qualification.

5. Section 4.5 - No credit should be given for "safe" failure modes in meeting this requirement. The comments of Section 4.4 apply. For example, if the most probable effect of a given accident is a loss of energy supply to the RTS, it does not matter, in meeting this requirement, whether or not the loss of energy causes the RTS to perform its protective function. Even though General Design Criterion 23 requires that the RTS be designed to "fail safe," acceptance of the RTS design shall not be based on an accident causing a failure, even if that accident-induced failure accomplishes the protective function.

6. Section 4.6 - The requirement for channel independence applies to all portions of the RTS that are designated as redundant channels. Independence is maintained in a number of ways. Physical independence is attained by physical separation and physical barriers. Electrical independence is achieved by isolation devices and utilization of separate power sources and other circuit devices. Verification of compliance with physical separation requirements may be made by comparing the design to Regulatory Guide 1.75 recommendations.

7. Section 4.7 - Control and protection system interaction involves more than examining their electrical isolation and interconnection. The functional performance of control systems must be reviewed to the extent that it is determined that a control system cannot prevent proper action of a protection system. This section of IEEE Std 279, with regard to isolation devices and multiple failures resulting from a credible single event, is explained by example in the document.

8. Section 4.8 - This requirement is self-explanatory. A protection system that requires loss of flow protection would normally derive its signal from flow sensors. A designer might elect to use an indirect parameter such as a pressure signal or pump speed. The reviewer should review the system to determine whether the indirect parameter would be valid at all times.

   Even a directly measured variable should be reviewed and its response to postulated events compared with the credit taken for the parameter in the events for which it provides protection.

9. Section 4.9 - The most common method used to verify the availability of the RTS input sensors is by cross checking between redundant channels that have readout available. When only two channels of readout are provided, evaluate the applicant's analysis of the effect of the operator choosing the incorrect readout as a basis for operator actions.

   When nonindicating sensors are used, check the test procedure to see whether a bypass indication is provided when the sensor is disabled. Of course, this latter approach should also be applied to indicating sensors when the design necessitates.

10. Section 4.10 - The extent of test and calibration capability that is provided bears heavily on whether the design meets the single failure criterion.

    a. Any failure that is not detectable must be considered concurrently with any postulated, detectable, single failure.

    b. Periodic testing should duplicate, as closely as practical, the overall performance required of the RTS. The test should confirm operability of both the automatic and manual circuitry. This capability must be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be reviewed in detail to confirm that the tests do, in fact, overlap from one test segment to another.

    c. Test frequencies are acceptable if identical to frequencies recently approved on other identical plants. Any changes made in the design or test procedures are not an adequate basis for reducing test frequencies until after experience is gained and the results submitted for review.

d. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation. Check that periodic tests conducted during power operation use only permanently installed test equipment. Also see Regulatory Guides 1.22 and 1.118 and Branch Technical Positions ICSB 22 and 25.

11. Section 4.11 - It is verified that tests can be conducted without initiating a protective action at the system level, and that tests can be conducted without preventing the initiation of a protective action at the system level. In general, it is an operational rather than a safety problem if testing causes the initiation of a protective action. For those parts of the RTS with a degree of redundancy greater than one, testing should not require bypass of the channel level protective action. For one-out-of-two systems, the channel protective action may be bypassed only if initiation of the protective action would disrupt plant operation. The bypassed channel must remain operable and operating. In these cases, verify that an interlock is provided that prevents, even with a single failure in the interlock circuits, bypassing both channels and that the single bypass is indicated. See Regulatory Guides 1.22 and 1.118.

12. Section 4.12 - The requirement for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may be required to take action to prevent the unnecessary initiation of a protective action and this is acceptable. In no circumstance should a design be approved where action or inaction of the reactor operator is required to make available the protective actions needed in any operational or shutdown mode of the plant.

13. Section 4.13 - See Regulatory Guide 1.47 and Branch Technical Position ICSB 21 for an explanation of this requirement as it pertains to the RTS.

14. Section 4.14 - In practice, administrative control is used as the basis for assuring that access to the means for bypassing is limited to qualified plant personnel and that permission of the control room operator is obtained to gain access.

15. Section 4.15 - This requirement is similar to Section 4.12. The phrase "positive means" can be interpreted as either automatic or manual. In the case of manual means, the design must be such that no action or inaction on the part of the reactor operator will prevent the more restrictive setpoint from being available. It is acceptable for the design to be such that incorrect action or inaction by the operator will cause an unnecessary protective action or prevent placing the plant in an operating mode for which there is inadequate protection.

16. Section 4.16 - "Completion of a protective action" must be defined by the applicant for the RTS. This information should be supplied as a part of the design basis information required by Section 3.0 of IEEE Std 279.

Generally, completion consists of causing negative reactivity to be inserted. Verify that once initiated, the protective action will continue to completion. Termination by deliberate actions of the operator should never inhibit the protective action.

17. Section 4.17 - Regulatory Guide 1.62 describes an acceptable method of implementing the requirement for manual initiation of protective actions. For those designs that take no credit (in the accident analyses) for manual initiation of protective actions, conformance with Regulatory Guide 1.62 is an adequate basis for acceptance. In practice, the requirements of IEEE Std 279 are applied to all equipment used by the operator to detect the need for the protective action, to accomplish the protection action, and to confirm completion of the protective action. However, it first should be established that automatic initiation need not or cannot be provided. Cost is not sufficient justification for the lack of automatic initiation.

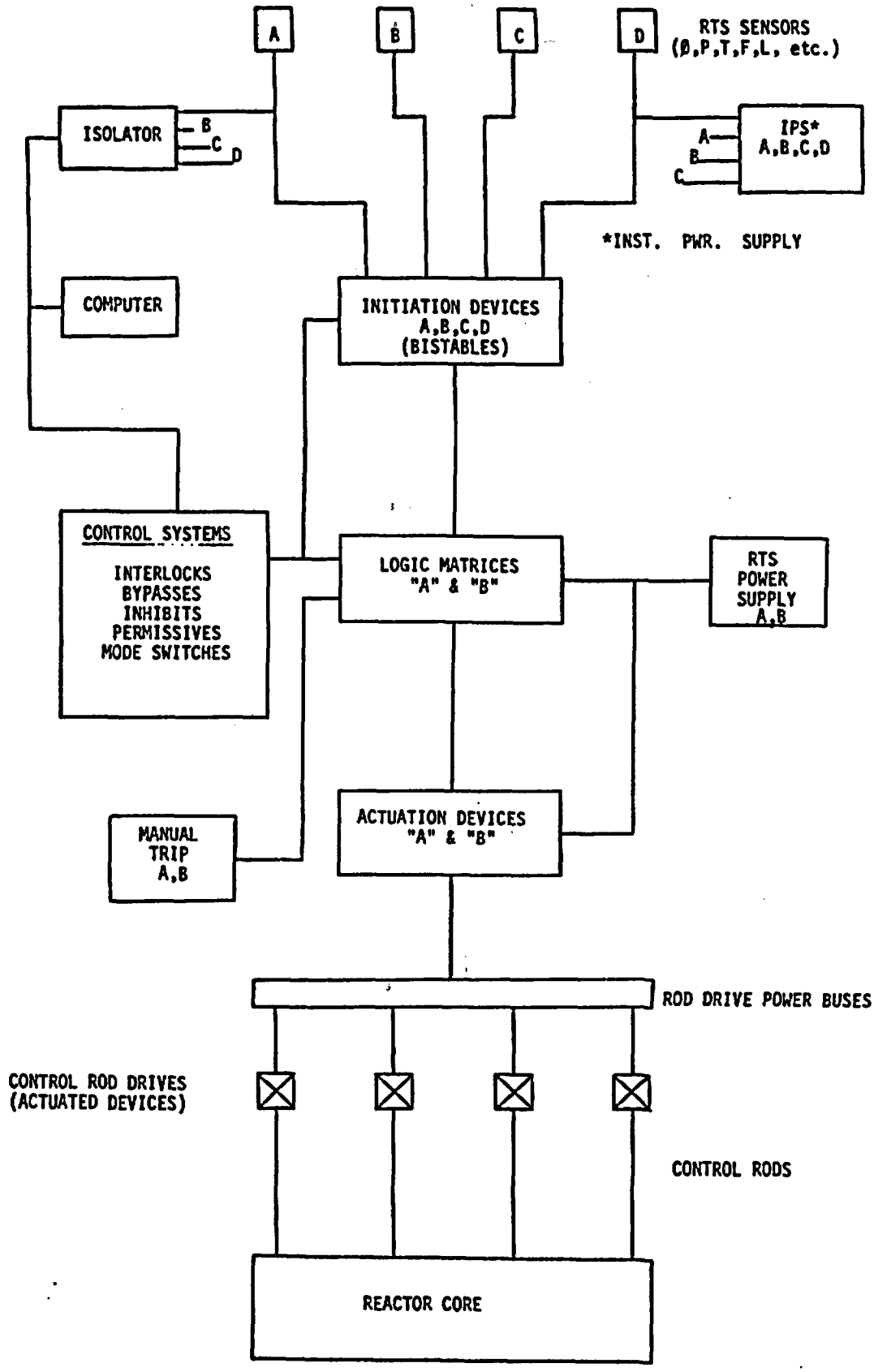
18. Section 4.18 - See procedure above for Section 4.14.

19. Section 4.19 - The method of identification of status at the channel level may be accomplished by lights, indicators, and annunciators.

20. Section 4.20 - The method used to establish adequacy of information readout would include a review of the RTS system inputs to annunciators and event recorders. Engineering judgment serves as the basis for acceptance.

21. Section 4.22 - This requirement is self-explanatory. The preferred identification method is color coding of components, cables, and cabinets. See also Regulatory Guide 1.75.

FIGURE 7.2-1
REACTOR TRIP SYSTEM
(Typical)



U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 7.3 ENGINEERED SAFETY FEATURES SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Auxiliary Systems Branch (ASB)
Containment Systems Branch (CSB)
Power Systems Branch (PSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

This SRP section describes the review for the portion of the protection system used to initiate and control operation of the engineered safety features systems and essential auxiliary supporting systems. This portion of the protection system is called the engineered safety features actuation system (ESFAS).

Typical engineered safety features (ESF) systems are:

Containment and Reactor Vessel Isolation Systems
Emergency Core Cooling Systems (ECCS)
Containment Heat Removal and Depressurization Systems
Pressurized Water Reactor (PWR) Auxiliary Feedwater Systems (see SRP Section 7.4 for review of the safe shutdown functions of this system)
Boiling Water Reactor (BWR) Standby Gas Treatment Systems
Containment Air Purification and Cleanup Systems
Containment Combustible Gas Control Systems

Typical essential auxiliary supporting systems are:

Electric Power Systems (see the SRP section in Chapter 8 for these systems)
Diesel Generator Fuel Storage and Transfer Systems
Instrument Air Systems
Heating, Ventilating, and Air Conditioning (HVAC) Systems for ESF Areas
Essential Service Water Systems

The descriptive information, functional control diagrams, piping and instrument diagrams, electrical schematics (operating license stage only), and physical arrangement drawings,

USNRC STANDARD REVIEW PLAN

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Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

as presented in the applicant's safety analysis report (SAR), are reviewed. The objectives are to determine that each engineered safety features actuation system satisfies applicable design criteria and will perform as intended during all plant operating conditions and accident conditions for which its function is required. The most significant difference between the review performed for a construction permit (CP) application and that performed for an operating license (OL) application is that the CP review can be based on a preliminary design. The depth of detailed information need only be "sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety" (Ref. 1). In addition, "a construction permit...will not constitute Commission approval of any design feature or specification with regard to safety unless the applicant specifically requests such approval and such approval is incorporated in the permit" (Ref. 2).

The review of the information presented and referenced in Section 7.3 of an SAR is primarily directed to the engineered safety features actuation system (ESFAS), i.e., the instrumentation and controls used to initiate and control the operation of the engineered safety features. The scope of the ICSB review of Section 7.3 of an SAR includes:

1. The descriptive information, including single line diagrams (CP) and schematic diagrams (OL) pertaining to the ESFAS. The ESFAS includes all electric and electromechanical equipment involved in detecting a plant condition requiring operation of an ESF system and in initiating the operation of the ESF system.

2. The descriptive information pertaining to the instrumentation and control systems for those auxiliary supporting systems that are essential to the operation of either the ESFAS or the ESF systems.

3. The applicant's proposed design criteria for the ESFAS and the instrumentation and controls of essential auxiliary supporting systems.

4. The applicant's analysis of the adequacy of the proposed design criteria and design bases for the ESFAS and the instrumentation and controls of auxiliary supporting systems.

5. The applicant's analyses of how the design of the ESFAS and auxiliary supporting systems conform to the design criteria for these systems.

The RSB and the CSB review, for those ESF systems within their review responsibilities, the following aspects of ESFAS:

(1) The adequacy of the monitored variables, i.e., the suitability of parameters, such as pressure, for initiating operation of a given ESF system.

(2) The acceptability of the proposed trip setpoints.

The PSB will assess the adequacy of physical separation criteria for cabling and electrical power equipment and determine that control and motive power supplied to redundant systems is from appropriate redundant sources. The PSB review includes the instrumentation and controls associated with the proper functioning of the onsite and offsite power systems. Detailed review procedures are set forth in Chapter 8.0 of the SRP.

The ASB will advise ICSB of any corrections to the SAR descriptions of auxiliary supporting systems essential to ESF systems and of time intervals available to initiate operation of auxiliary supporting systems.

II. ACCEPTANCE CRITERIA

Acceptance criteria for the review areas of this SRP section are referenced in Table 7-1 (Ref. 3), and include the General Design Criteria, industry standards, regulatory guides, and branch technical positions that are applicable to the ESFAS and the instrumentation and controls of essential auxiliary supporting systems. These documents either establish design requirements or describe acceptable methods of implementing design requirements. In each of these categories, some documents set forth mandatory design criteria and others describe acceptable methods of design.

The General Design Criteria and IEEE Std 279 set forth requirements that must be met by all designs for the ESFAS. In addition, these are also used for the instrumentation and controls for the essential auxiliary supporting systems. One purpose of the review is to verify that the applicant has committed to designing the ESFAS and the essential auxiliary supporting system instrumentation and controls in accordance with these mandatory criteria.

The regulatory guides are not mandatory and only set forth acceptable methods of implementing the mandatory criteria. The branch technical positions are used when a particular design problem has an identified and acceptable solution; they also are not mandatory.

Industry standards that are not endorsed by regulatory guides or incorporated in regulations or technical positions, or that have not been previously used and accepted in the licensing process, must be reviewed before they can be accepted as a sole basis for approval of a design. They are useful as guidance for identifying the subjects of importance to be considered in the review of the ESFAS. In all cases, the primary basis for acceptance of an ESFAS design is conformance to the mandatory criteria of the regulations.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

This subsection describes the general procedures to be followed in reviewing the ESFAS. For simplicity, it is written for the ESFAS for a single ESF system comprised of two identical, redundant subsystems. The same procedure should be applied to each ESF system and to each essential auxiliary supporting system.

Background information of interest in the review of the ESFAS is found in a number of SAR sections. A list of these is given below for reference purposes. Most of these reference sections also provide background information for other SRP sections in Chapter 7.

Chapter 1: for familiarization with the general operation of the plant, both safety and nonsafety aspects.

Chapter 3: for a general understanding of the principal architectural and engineering designs of those structures, components, equipment, and systems important to safety.

Section 3.1: for exceptions to criteria applicable to the ESFAS, and for structures suitable for housing ESFAS equipment.

Chapters 4 and 5: for an understanding of the reactor and the reactor coolant system and its interconnections with the ESF systems.

Chapter 6: for the design bases, design features, and functional performance requirements of the ESF system.

Chapter 7: for a detailed understanding of the design and operation of the ESFAS.

Chapter 9: for the design bases, design features, and functional performance requirements of essential auxiliary supporting systems.

Chapter 15: for the courses of accidents for which the ESF system provides protective functions, the effects of failures of the protective functions, and the assumptions and initial conditions that form the bases of the accident analyses.

Chapter 16: for the proposed limiting conditions for operation for the ESF and the ESFAS.

It should be noted that reference to the above sections of the SAR is made to gain an understanding of the purpose of the ESF and an understanding of how the ESF system and the ESFAS are designed and are supposed to function. No "evaluation" should be made of these sections, i.e., the SAR description is taken at face value.

The next step is to evaluate the design against the requirements of IEEE Std 279. This procedure is detailed in Appendix A to this SRP section. The procedures in Appendix A address only those design requirements that are specific in nature. For example, paragraph 4.9 of IEEE Std 279 requires that the design include means for checking the availability of each system input sensor during operation. Appendix A outlines a straightforward procedure that can be used to determine whether or not this requirement is met.

Appendix A discusses the requirements of IEEE Std 279 and how they are used in the review of the ESFAS and the essential auxiliary supporting systems instrumentation and controls. Although the primary emphasis is on the equipment comprising the ESFAS, the reviewer should consider the protective functions on a systems level. It is necessary that the ESFAS

design be compatible with the ESF systems and auxiliary supporting systems and that the ESFAS design and the accident analysis are compatible. It is not sufficient to judge the adequacy of the ESFAS only on the basis of the design meeting the specific requirements of IEEE Std 279. It is also necessary to judge the functional relationship between the ESFAS and the ESF systems themselves.

Other requirements for the ESFAS and the instrumentation and controls of essential auxiliary supporting systems are listed in Table 7-1. Many of these requirements are general in nature, permitting the acceptability of various designs. For example, General Design Criterion 20 requires, in part, that the protection system be designed to sense accident conditions and to initiate the operation of (ESF) systems important to safety. A cursory examination of the descriptive information would be sufficient to determine whether or not the ESFAS is designed to sense accident conditions and initiate the ESF systems. The specific review procedures for such general requirements are not detailed here. Specific design features and approaches are described in the ICSB technical positions in Appendix 7-A to Chapter 7 of the SRP.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

In certain instances, it will be the reviewer's judgment that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a nonuniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"7.3 Engineered Safety Feature Actuation Systems (ESFAS)

"The engineered safety features actuation systems include the instrumentation and controls used to detect a plant condition requiring operation of an engineered safety features (ESF) system, to initiate action of the ESF, and to control its operation. The scope of review of the ESFAS for the _____ plant included single line diagrams (CP and OL) and schematic diagrams (OL) and descriptive information for the ESFAS and for those auxiliary supporting systems that are essential to the operation of either the ESFAS or the engineered safety features systems themselves. The review has included the applicant's proposed design criteria and design bases for the ESFAS and the instrumentation and controls of auxiliary supporting systems, and his analysis of the adequacy of those criteria and bases. The review also has included the applicant's analyses of the manner in which the design of the ESFAS and the auxiliary supporting systems conform to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the engineered safety features actuation systems and necessary auxiliary supporting systems to the Commission's regulations as set forth in General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

"The staff concludes that the design of the engineered safety features actuation systems conform to all applicable regulations, guides, branch technical positions, and industry standards and is acceptable."

V. REFERENCES

1. 10 CFR §50.34(a)(3)(iii), "Contents of Applications: Technical Information; Preliminary Safety Analysis Report."

2. 10 CFR §50.35(b), "Issuance of Construction Permits."

3. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."

4. Standard Review Plan Appendix 7-A, "Branch Technical Positions (ICSB)."

APPENDIX A

STANDARD REVIEW PLAN SECTION 7.3

USE OF IEEE Std 279 IN THE REVIEW OF THE ESFAS AND INSTRUMENTATION AND CONTROLS OF ESSENTIAL AUXILIARY SUPPORTING SYSTEMS

This appendix discusses the requirements of IEEE Std 279, Sections 3 and 4, as they are used in the review of the ESFAS and instrumentation and controls of essential auxiliary supporting systems.

1. Section 3 - This section requires that a specific protection system design basis be provided for each protection system. (A protection system is defined by General Design Criterion 20.)

   Much of the material in Section 3 is stated as requirements in Section 4. However, extra attention should be paid to Section 3(9), minimum performance requirements, because this has been historically a problem area.

   a. System response times - The maximum and/or minimum response times must be stated so that compliance with IEEE Std 279, Section 4.10, can be assured.

   b. System accuracies - See Regulatory Guide 1.105.

   c. Ranges - See Regulatory Guides 1.29, 1.89 and 1.105.

2. Section 4.1 - This section requires that the ESFAS perform automatically and with precision and reliability. These requirements must be met over the full range of transient and steady-state conditions of the energy supply and environment during all plant conditions in which the applicant's accident analyses take credit for functions performed by the ESFAS. Other criteria which set forth similar requirements are General Design Criteria 2, 4, 10, 13, 20, 21 and 29.

   a. Automatic initiation is required for all protective functions that must be started within a short time of the indicated need for the function. Although General Design Criterion 20 appears to require automatic initiation of all protective functions, initiation solely by manual means has been acceptable. However, automatic initiation is preferable for all protective functions, even though they are not needed (according to the accident analyses) for a relatively long time. Where the protective action is initiated solely by manual means, all the actions that need or may need to be performed by the operator during the time interval are reviewed, as are the applicant's basis for not providing automatic initiation. In this latter regard, the cost of automatic initiation is not, of itself, sufficient justification for using manual initiation. If in the reviewer's judgment manual initiation is sufficiently reliable, then the equipment used by the operator to detect the need for the protection function, and to verify that the protective function has been completed, must also meet all the

requirements applicable to automatically initiated protective functions. See also Branch Technical Position (BTP) ICSB 20.

b. The precision required in the ESFAS is at least that assumed in the accident analyses.

c. There are no quantitative requirements established for the reliability of the ESFAS. The design is reviewed to identify any unusual or unique equipment that has not previously been used in nuclear plants. The "type testing" (as defined in IEEE Std 323) that demonstrates such equipment is capable of performing its function is reviewed. The design is also reviewed to assure that no unnecessary interlocks, time delays, or other complexities are introduced in the ESFAS circuits. Where such features do exist, the applicant's design bases and performance analyses should be reviewed to determine that the reliability of the ESFAS is not significantly reduced by the inclusion of such features.

3. Section 4.2 - This is the most fundamental of all the requirements that the ESFAS must meet. It is inherent in other criteria such as General Design Criteria 21, 22, 24, 34, 35, 38, 41, 44, 54, 55 and 56.

In evaluating ESFAS conformance with this requirement, the reviewer must examine several different aspects of each single failure to determine its effect. The time of occurrence of the failure and the plant conditions prevailing at that time can significantly alter the effects of any single failure.

a. The first step in a single failure analysis is to identify components in the ESFAS that are not seismic Category I, those that are not qualified for accident and post-accident environments, and those that serve both safety and nonsafety systems and whose failure can affect the performance of or create the need for the EFSAS. Each of the nonqualified and nonsafety-grade systems and components are assumed to fail to function if failure adversely affects ESFAS performance and are assumed to function if functioning adversely affects ESFAS performance. However, multiple independent failures are not assumed to occur simultaneously.

b. Next, the consequences of the events for which the ESFAS is designed to provide protective functions are examined. All failures that can be predicted to occur as a direct or consequential result of an event are assumed to occur if such failures adversely affect ESFAS performance. In general, lack of adequate environmental or seismic qualification testing is sufficient basis to assume a direct or consequential failure of equipment.

c. After assuming the failures of nonsafety-grade, nonqualified equipment and those failures caused by an event, any other single failure in the ESFAS or its auxiliary supporting systems is arbitrarily assumed and the resultant performance of the ESFAS is analyzed to assure that sufficient equipment is available to perform the minimum protective function.

d. In choosing the postulated failure to be analyzed, no distinction is made between active and passive components in electrical systems. Further, electrical equipment serving mechanical components that are not required to function in a given event is treated the same as electrical equipment serving "active" mechanical components, i.e., those that must function. (See also BTP ICSB 18 (PSB).)

e. The meaning of redundancy is discussed in IEEE Std 379 and Regulatory Guide 1.53. Basically, to be considered redundant there must be no communication, either directly or indirectly, between two systems that can perform the same function. Thus, two systems, each of which can perform a protective function, are not redundant (and therefore do not meet the single failure criterion) if the failure of one system affects in any way the performance of the other system. This includes starting (or not starting) one system by sensing the failure (or operation) of the other system.

4. Section 4.3 - There are at present no specific criteria to judge the quality of the equipment used in the ESFAS. However, Appendix B to 10 CFR Part 50 provides some guidance from which a judgment may be made of the quality of equipment required for the ESFAS.

5. Section 4.4 - SRP Sections 3.10 and 3.11 discuss the evaluation of equipment qualification. In reviewing the ESFAS, check that each component or module of the ESFAS has been qualified for normal, accident, and post-accident environments at its installed location. This applies to all normal conditions but only to those accident conditions where the component or module provides a protective function.

6. Section 4.5 - This requirement is similar to Section 4.4 discussed above. No credit should be given for "safe" failure modes in meeting this requirement. For example, if the most probable effect of a given accident is a loss of energy supply to an ESFAS, it does not matter, in meeting this requirement, whether or not the loss of energy causes the ESFAS to perform its protective function. Even though General Design Criterion 23 requires that the ESFAS be designed to "fail-safe," acceptance of the ESFAS design should not be based on an accident causing a failure, even if that accident-induced failure accomplishes the protective function.

7. Section 4.6 - The requirement for channel independence applies to all portions of the ESFAS that are designated as redundant channels. Verification of compliance with this requirement and the recommendations of Regulatory Guide 1.75 and IEEE Std 384 concentrates on points of interface between redundant ESFAS components and interfaces between the redundant portions of the ESFAS and nonsafety-grade systems. For example, switches common to redundant portions of the ESFAS are reviewed for physical independence between redundant switch sections and for the effects on redundant systems caused by a single malpositioned switch. Also reviewed are the functional performances of isolation devices to assure that no failure in nonsafety circuits can disable safety functions.

8. Section 4.7 - The interaction of control systems and the ESFAS involves more than examining the electrical interconnection of control systems with the ESFAS. Compliance with the diversity requirements of subsection 4.7.4.1 is a requirement for the initiation of

engineered safety features and the interlocks for valves between the reactor coolant system and low pressure systems. In addition, the functional performance of appropriate control systems must also be reviewed to determine whether their effect on plant conditions can indirectly affect the performance of the ESFAS or the ESF. For example, if a cooling water system is used to supply both safety and nonsafety equipment, the controls for the cooling water system must be examined to determine whether failure could lead to insufficient cooling water being supplied to the ESF or the ESFAS during an accident. (Also see Regulatory Guide 1.106.)

Note that if failure of a system serving both safety and nonsafety systems can lead to a condition requiring action by the safety system, then in addition to the failure creating the need for safety action, the ESFAS must be designed to withstand any other simultaneous single failure.

9. Section 4.8 - This requirement is self-explanatory. In addition, it must be verified that the measured variable is the variable that is used in the accident analyses.

10. Section 4.9 - The most common method used to verify the availability of the ESFAS input sensors is by cross checking between redundant channels that have readout available. When only two channels of readout are provided, the applicant's analysis of the effect of the operator choosing the incorrect readout must be evaluated as a basis for this action.

    Where non-indicating sensors are used, check the test procedure to see whether a bypass indication is provided when the sensor is disconnected from the process system.

11. Section 4.10 - The extent of test and calibration capability that is provided bears heavily on whether the design meets the single failure criterion.

    a. Any failure that is not detectable must be considered concurrently with any postulated, detectable, single failure.

    b. Periodic testing should duplicate, as closely as practical, the integrated performance required from the ESFAS, ESF systems, and their essential auxiliary supporting systems. If such a "system level" test can be performed only during shutdown, the testing done during power operation must be reviewed in detail. Check that "overlapping" tests do, in fact, overlap from one test segment to another. For example, closing a circuit breaker with the manual breaker control switch may not be adequate to test the ability of the ESFAS to close the breaker.

    c. Test frequencies are acceptable if identical to frequencies recently approved on other identical plants. Any changes made in design or test procedure are not an adequate basis for reducing test frequencies until after experience is gained and the results submitted for review. For new designs, test frequencies should conform to Section 6.5 of IEEE Std 338.

d. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation. Check that periodic tests conducted during power operation use only permanently installed test equipment. See also Regulatory Guides 1.22 and 1.118, BTP ICSB 22 and 25, and IEEE Std 338.

12. Section 4.11 - Verify that tests can be conducted without initiating a protective action at the system level, and that tests can be conducted without preventing the initiation of a protective action at the system level. In general, it is an operational rather than a safety problem if testing causes the initiation of a protective action. For those parts of the ESFAS with a degree of redundancy greater than one, testing should not result in a loss of ESFAS function even if a single failure exists in one of the redundant channels which are not under test. For one-out-of-two systems, one channel may be bypassed only if initiation of the protective action would disrupt plant operation and the other channel remains operable. In these cases, verify that an interlock is provided that prevents, even with a single failure in the interlock circuits, bypassing both channels and that the single bypass is indicated. See also Regulatory Guides 1.22 and 1.118, and IEEE Std 333.

13. Section 4.12 - The requirement for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may be required to take action to prevent the unnecessary initiation of a protective action and this is acceptable. Under no circumstances should a design be approved where action of the reactor operator is required to make available the protective actions needed in any operational or shutdown mode of the plant.

14. Section 4.13 - See Regulatory Guide 1.47 and BTP ICSB 21 for an explanation of this requirement as it pertains to the ESFAS, ESF systems, and auxiliary supporting systems.

15. Section 4.14 - In practice, administrative control is used as the basis for assuring that access to the means for bypassing is limited to qualified plant personnel and that permission of the control room operator is obtained to gain access.

16. Section 4.15 - This requirement is similar to Section 4.12. The phrase "positive means" can be interpreted as either automatic or manual. In the case of manual means, the design must be such that inaction on the part of the reactor operator will not prevent the more restrictive setpoint from being available when it is required. It is acceptable for the design to be such that incorrect action or inaction by the operator will cause an unnecessary protective action or prevent placing the plant in an operating mode for which there is inadequate protection (as defined by the accident analyses). See BTP ICSB 12 for specific guidance on setpoint changes required with a reactor coolant pump out of service.

17. Section 4.16 - For the ESFAS, "completion of a protective action" must be defined by the applicant for each ESF system. This information should be supplied as part of the design basis information required by Section 3.0 of IEEE Std 279.

Generally, completion consists of starting or energizing the components in the ESF system. Verify that once initiated, the protective action will continue until terminated by deliberate actions of the operator and that operator action cannot prevent the initiation of the protective action when the ESFAS determines the need for that action. Exception: "pull-to-lock" control switches have been acceptable even though their manipulation could prevent the protective action from going to completion provided that the requirements of Section 4.13 are met.

18. Section 4.17 - Regulatory Guide 1.62 describes an acceptable method of implementing the requirement for manual initiation of protective actions. For those designs that take no credit (in the accident analysis) for manual initiation of protective actions, conformance with Regulatory Guide 1.62 is an adequate basis for acceptance.

    For those protective actions which are initiated solely by manual means, there are no specific criteria to judge acceptance at present. In practice, the requirements of IEEE 279 are applied to all equipment used by the operator to detect the need for the protective action, to accomplish the protection action, and to confirm completion of the protective actions. However, it first should be established that automatic initiation need not or cannot be provided. Cost is not sufficient justification for the lack of automatic initiation. In judging the adequacy of any manual initiation features, the other tasks that the operator may be required to perform should be determined and then a judgment made as to whether it is reasonable to rely on the operator to perform all necessary actions. In most situations automatic actuation, backed up by provisions for manual initiation or manual termination, is more reliable than manual initiation alone, no matter how much time is available to take the protective action.

19. Section 4.18 - See procedure above for Section 4.14.

20. Sections 4.19 and 4.20 - Other than the requirements for indication and identification of channel and system level protective actions, there are no specific implementation guidelines by which to judge the adequacy of a design with respect to the requirements for status indication. Evaluate the applicant's discussion of how the ESFAS designs conforms to these requirements. Acceptance is based on the reviewers's engineering judgment.

    See also SRP Section 7.5 for a discussion of review procedures for safety-related display instrumentation.

21. Section 4.22 - This requirement is self-explanatory. The preferred identification method is color-coding of components, cables, and cabinets. See also Regulatory Guide 1.75.

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SECTION 7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Reactor Systems Branch (RSB)
Containment Systems Branch (CSB)
Auxiliary Systems Branch (ASB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)
Power System Branch (PSB)

I. AREAS OF REVIEW

The systems reviewed under this SRP section are those instrumentation and control systems associated with parts of the nuclear steam supply system (NSSS) used to achieve and maintain a safe shutdown condition of the plant. The specific arrangement of these parts of both the primary and secondary loops of the NSSS depends on the type of plant (pressurized water reactor, PWR; boiling water reactor, BWR; etc.) as well as on individual plant design features, and the conditions under which the safe shutdown has to be achieved and maintained. There are two kinds of shutdown conditions: hot shutdown and cold shutdown. In either case, it is necessary that reactivity control systems maintain a subcritical condition of the core and that residual heat removal systems operate to maintain adequate cooling of the core. For a precise definition of both shutdown conditions for a specific plant, see Chapter 16, "Technical Specifications," in the applicant's safety analysis report (SAR).

Examples of systems required for achieving and maintaining a safe shutdown are the auxiliary feedwater system, the residual heat removal system, and the boric acid transfer system (for PWRs).

The review of the instrumentation and control systems associated with the various parts of the NSSS required for safe shutdown, along with the equipment required for their proper alignment from the main control room or from other locations outside the control room, is the responsibility of the ICSB. The review includes the sensors, initiating circuitry, logic bypasses, interlocks, redundancy features, and actuated devices of those systems and auxiliaries which provide the necessary instrumentation and control functions. The ICSB review should confirm that the systems required for safe shutdown, as defined above, and their supporting systems will perform design functions as required for plant shutdown and conform to all applicable acceptance criteria.

USNRC STANDARD REVIEW PLAN

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The descriptive information contained in the SAR, including single line diagrams, electrical schematics, piping and instrumentation diagrams (P&IDs), and physical arrangement diagrams are reviewed to ascertain that the systems required for safe shutdown meet the acceptance criteria listed in subsection II. For a construction permit (CP) review, a commitment to meet these criteria, together with a preliminary design, can suffice in cases where the design of these systems has not been completed. For an operating license (OL) review, however, the actual design and its implementation should be verified to meet these criteria.

The ICSB review includes the following specific items:

1. The redundancy of logic and instrumentation provided for the operation and status monitoring of systems required for safe shutdown. This requires the review of the descriptive information contained in the SAR, functional diagrams, electrical schematics, and P&IDs.

2. The ability of systems required for safe shutdown to function after sustaining a single failure. This requires the review of the descriptive information and diagrams as in (1), above, and in addition the drawings showing the physical layout of the instrumentation and control equipment and their associated cabling.

3. The criteria for design of instrumentation and structures housing parts of the systems required for safe shutdown.

4. The environmental qualification of the electrical and instrumentation equipment and cabling.

5. The online testability of the systems and indication of bypassed or inoperable status of the systems required for safe shutdown.

The PSB evaluates the redundancy of power sources, the design criteria for physical separation of redundant electrical equipment, cabling, and cable trays, the design criteria for providing control and motive power to these systems, the control arrangements for manually-controlled electrically-operated valves, and provisions for sharing of electrical systems between units in multi-unit plants.

The ASB should evaluate the adequacy of those auxiliary systems required for the proper operation of the systems required for safe shutdown. These include systems concerned with compressed air requirements, reactor coolant chemistry, boron concentration, lighting, air conditioning, etc. In particular, the ASB should determine that the piping, ducting, and valving of redundant vital auxiliary supporting systems meet the single failure criterion. In addition, the ASB should review the physical arrangement of components and structures related to the systems required for safe shutdown and their supporting systems, and determine that single events will not disable these systems.

The CSB should review the containment ventilation and atmosphere control systems provided to maintain required environmental conditions for electrical and instrumentation equipment associated with the systems required for safe shutdown and located inside containment.

The MEB review should confirm that the seismic qualification of instrumentation and electrical systems is acceptable. This includes the design criteria and testing methods and procedures employed in the seismic design and installation of Category I instrumentation and electrical equipment.

The RSB should review the systems identified as required for safe shutdown, and confirm that the configuration and design bases of these systems are correct, and that all design parameters such as temperature, pressure, flow rate, and reactivity can be controlled within acceptable limits. This information should be provided to the ICSB. For situations where shutdown is to be accomplished from locations outside of the main control room, the RSB review should establish the adequacy of needed systems and any differences in system alignment or operation that are required to achieve and maintain safe shutdown.

The QAB review should verify that the quality assurance program proposed by the applicant includes the systems required for safe shutdown.

II. ACCEPTANCE CRITERIA

The design materials, qualifications testing, and surveillance of systems required for safe shutdown are covered by several General Design Criteria (GDC), IEEE standards, regulatory guides, and branch technical positions. A list of applicable criteria, standards, guides, and technical positions is given in Table 7-1 and Appendix 7-A, attached to the SRP section for Chapter 7.

The instrumentation and control systems required for safe shutdown are acceptable when it is determined that these systems satisfy the following requirements:

1. They have the required redundancy.

2. They meet the single failure criterion.

3. They have the required capacity and reliability to perform intended safety functions on demand.

4. They are capable of functioning during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.

5. They are testable during reactor operation.

The criteria listed in Table 7-1 are utilized as the bases for determining that these requirements are met. How these criteria are applied during the review process is discussed in subsection III. The applicability of the acceptance criteria to the review of the systems required for safe shutdown is as follows:

1. System Redundancy Requirements

   General Design Criteria 26, 33 and 34, and IEEE Std 279 specify the requirements that systems required for safe shutdown, among others, must meet with regard to all operating conditions (such as loss of offsite power), so that they can perform their safety function assuming a single failure. If a determination is made that the systems required for safe shutdown meet the requirements of these criteria, they are acceptable in this regard. Electrical and physical independence requirements as discussed in SRP Sections 7.2 and 7.3 should be met.

2. Conformance with the Single Failure Criterion

   IEEE Std 279, IEEE Std 379, and Regulatory Guide 1.53 provide recommendations and guidance for meeting the single failure criterion. Regarding the application of the single failure criterion to the design of manually-controlled electrically-operated valves, the acceptability of proposed designs is based on Branch Technical Position ICSB 18.

3. Identification of Cables, Cable Trays, and Instrument Panels

   The method used for identifying power and signal cables and cable trays as safety-related equipment, and the identification scheme used to distinguish between redundant cables, cable trays, and instrument panels should be in accordance with the recommendations of Sections 5.1.2 and 5.6.3 of Regulatory Guide 1.75, "Physical Independence of Electric Systems," and Section 4.2.2 of IEEE Std 279. Color coding is a preferred method of identification.

4. Vital Supporting Systems

   The instrumentation, control, and electric equipment associated with the auxiliary systems that support the systems required for safe shutdown should meet the same acceptance criteria as for the systems they support.

5. System Testing, Quality Assurance, and Surveillance

   General Design Criteria 1 and 21, IEEE Std 279, IEEE Std 336, and Regulatory Guides 1.22, 1.47 and 1.68 contain the applicable acceptance criteria with regard to preoperation and periodic testing, quality assurance, and design provisions for indicating the availability of systems required for safe shutdown and essential auxiliary supporting systems.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The main objectives of the review of systems required for safe shutdown are to determine that the design of these systems includes the required redundancy; meets the single failure criterion; provides the required capacity and reliability to perform intended safety functions on demand; and provides the capability to function during and after design basis events such as earthquakes and anticipated operational occurrences.

For a CP review, the descriptive information contained in the preliminary safety analysis report (PSAR), including the design bases and their justification with regard to the acceptance criteria, electrical single line drawings, and P&IDs, are reviewed to determine that the basic design features and the commitments made provide assurance that the final design will meet the acceptance criteria. During the OL stage of review it is verified that the acceptance criteria are met through review of the final electrical and instrumentation drawings and the physical layout drawings, and a site visit, during which a spot-check verification of the design is performed. In order to verify that the acceptance criteria are satisfied, the review is performed in accordance with the following specific procedures.

A major portion of the systems required for safe shutdown are also used as engineered safety features systems, as discussed in SRP Section 7.3. Therefore, the review under this SRP section includes the safe shutdown systems configurations which result from a realignment of ESF systems in addition to the safe shutdown systems configurations which are not part of engineered safety features systems. The RSB and ASB confirm the acceptability of the proposed configuration and the redundancy required for systems required for safe shutdown as specified in General Design Criteria 26, 33 and 34. The descriptive information, including the electrical one-line diagrams and P&IDs (for CP and OL reviews) and electrical schematics (for the OL review), should be reviewed to verify that the necessary redundancy is provided. This should include instrumentation channels used to sense vital parameters such as temperature, pressure, water level, etc.; the associated logic and actuated devices; and the motive and control power sources.

Conformance with the single failure criterion as specified by IEEE Std 279 and Regulatory Guide 1.53 is verified by review of the same information as for redundancy and may be done, to some extent by necessity, at the same time. The guidance provided by Regulatory Guide 1.53 is excellent for ascertaining that a given design is single failure proof. A particularly important but subtle point to check is one cited in position 4 of Regulatory Guide 1.53, wherein a single d-c source supplies control power for one channel of system logic and for the redundant actuator circuit.

Certain areas of review need close coordination between primary and secondary review branches in order to make a determination that a specific aspect of the design meets the applicable criteria. Seismic qualification of Class 1E equipment, flood protection of safety-related systems and components, and effects of high energy fluid line breaks inside containment or near safety-related equipment are the major areas for which branch coordination is essential in evaluating the acceptability of a given design feature.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

For a multi-unit plant where electrical systems are shared, thus resulting in more and complex interaction modes, a fault-tree and decision-tree analysis may be requested from the applicant to show that a single failure or a single event resulting in multiple failures will not result in unacceptable consequences with respect to the capability of systems required for safe shutdown to perform safety functions when required. Additional guidance with regard to the single failure criterion as it relates to shared electric power systems is given in the SRP sections for Chapter 8.

For the case of manually-controlled electrically-operated valves in these systems, the acceptability of the proposed design is based on satisfying Branch Technical Position ICSB 18 (PSB). This position basically states that it is acceptable to disconnect electric power to a safety-related valve as means of removing the possibility of an active failure of that valve.

Regulatory Guide 1.75 provides guidance for satisfying the acceptance criteria with respect to the identification of power and signal cables, cable trays, and instrument panels related to systems required for safe shutdown. The criteria for identification and separation of redundant systems discussed in Regulatory Guide 1.75 are presented in sufficient detail to make their application self-explanatory.

General Design Criteria 1, 21, 22 and 23, IEEE Std 279, IEEE Std 336, and Regulatory Guides 1.22, 1.47 and 1.68 provide the requirements that the design of systems required for safe shutdown must meet with regard to preoperational and periodic inservice testing. The primary review responsibility for the preoperational testing is with the QAB. Periodic testing and downtime restrictions are specified in the technical specifications. The review procedures for technical specifications are covered in SRP Section 7.1.

Another important area to be reviewed is the remote or local control stations that are required by General Design Criterion 19 for the safe shutdown of the plant in case the main control room becomes uninhabitable. Plant designs should provide for control stations in locations removed from the main control room that may be used for manual control and alignment operations needed to achieve and maintain a hot shutdown and subsequently to be able to achieve a cold shutdown. Equipment required for safe shutdown should be operable from local control panels. Access to these local control panels should be under strict administrative controls. The design of these control stations should provide appropriate readouts so that the operator can monitor the status of the shutdown. Typical readouts are steam generator level, steam generator pressure, pressurizer pressure, pressurizer level, and auxiliary feedwater flow.

The remote control stations and the equipment used to maintain safe shutdown should be designed to accommodate a single failure. Equipment located at these stations which is required for safe shutdown should be capable of operating independently (without interaction) of the equipment in the main control room. The design should be such as to prevent a single failure in the main control room or the cable spreading room from defeating the capability for affecting safe shutdown from the remote control stations, and vice versa. The remote control station equipment should be designed to the same standards as the corresponding equipment in the main control room, including appropriate IEEE criteria. Control transfer devices should be located away from the main control room and cable spreading areas, and their actuation should cause an alarm in the control room.

An important part of the review is the engineering drawing review at the OL stage. The drawing review should:

1. Verify that a complete set of drawings has been submitted that includes logic diagrams, P&IDs, and location layout drawings for these systems.

2. Verify that the submitted drawings represent the actual system designs and layouts for the particular plant, and that those drawings submitted as "typical" of a system are so identified.

3. Verify that the design and layout meet the applicable criteria listed under subsection II.

The environmental qualification of components and cabling of systems required for safe shutdown should be the same as for the ESF systems discussed in SRP Section 7.3.

An applicant may choose to take exceptions to some of the acceptance criteria in the branch technical positions, guides, IEEE standards (other than IEEE Std 279, which is a mandatory requirement) and propose alternate ways of meeting the General Design Criteria requirements (which are mandatory). Any exceptions to the criteria are evaluated on an individual case basis. Exceptions are judged on the basis of the proposed design providing an equivalent level of safety and conservatism.

In general, the applicant will have design criteria that supplement or clarify the mandatory criteria. In the evaluation of such criteria, the reviewer can use the guidance listed above to determine whether the applicant's design criteria are adequate.

For the purpose of the ICSB review, no distinction should be made between the design criteria for systems required for safe shutdown and the criteria for the instrumentation and controls for essential auxiliary supporting systems.

Certain system designs and design features are submitted on a generic basis in the form of topical reports. Reference to a topical report is an acceptable alternative to

submitting information in an application for a CP or an OL. Generally, topical reports pertain to standardized systems and qualification tests. If a referenced topical report has been accepted after staff review, the subjects of the report should not be reviewed again in connection with a particular application. If the referenced topical report has not been reviewed up to the time of the application review, it should be reviewed and treated in the same manner as the SAR itself. It may be necessary to assure conformance to requirements by getting additional information and justification from the applicant. If the topical report has been rejected, then the applicant should be so advised and requested to submit information or design changes that are acceptable.

References other than topical reports should be obtained from the library or other sources, or the applicant asked to supply a copy.

A site visit and inspection should be performed before the evaluation findings are written for an OL. A site inspection should include a spot-check verification that the design and layout criteria reviewed during the drawing review are actually implemented at the hardware assembly stage. A site visit should be coordinated with the licensing project manager and the regional office that has jurisdiction over the geographic area in which the plant is located. Specific items to be considered include:

1. Separation and identification of redundant safety-related instrumentation channels, cabling, cable trays, and instrument rack terminations.

2. Separation of actuating switches in control panels for redundant safety-related equipment such as inboard and outboard isolation valves, coolant pumps, diesel generator sets, etc.

3. Testability provisions and calibration procedures for instrumentation channels required for safe shutdown.

4. Adequacy of local control panels for remote shutdown, especially with regard to sufficient monitoring channels and actuating devices that the operators would need to perform and maintain a safe shutdown.

For a full outline of topics for a site visit, see Appendix 7-B to Chapter 7 of this SRP.

In certain instances, it will be the reviewer's judgment that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been submitted and the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The review of systems required for safe shutdown includes the sensors, initiating circuitry, logic elements, interlocks, redundancy features, actuated devices, and auxiliaries that provide the instrumentation and control functions that prevent the reactor from returning to criticality and provide means for adequate residual heat removal from the core, containment, and other vital components and systems.

"The scope of review of systems required for safe shutdown for the plant included single line diagrams (CP and OL) and schematic diagrams (OL) and descriptive information for these systems and for auxiliary systems essential for their operation. The review has included the applicant's proposed design criteria, design bases, and analyses. The review has also included the applicant's analyses of the manner in which the design of these systems and their auxiliary supporting systems conform to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for systems required for safe shutdown and essential supporting auxiliaries to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

"The staff concludes that the design of systems required for safe shutdown conforms to the applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (ICSB)."

3. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

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SECTION 7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Reactor Systems Branch (RSB)
Core Performance Branch (CPB)
Auxiliary Systems Branch (ASB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)
Containment Systems Branch (CSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

Information presented in the applicant's safety analysis report (SAR) is reviewed by the staff to determine that the design of safety-related display instrumentation (SRDI) required for safe functioning of the plant during operating and accident conditions is in conformance with applicable regulations, guides, branch technical positions, and industry standards and is consistent with the accident analysis assumptions of Chapter 15 of the SAR. For construction permit (CP) applications, the applicant's descriptive information for the SRDI should include commitments to meet applicable requirements and should present full justification for any exceptions taken.

For operating license (OL) reviews, the information presented should include the following:

1. Tables of system variables and components to be indicated and recorded (including accuracies and ranges of instruments).

2. Functional control diagrams or other means of illustrating the redundancy of monitored variable and component sensors and channels, the capability for sensor checks, and the means for verifying operability of monitoring system channels.

3. Electrical distribution diagrams illustrating electrical isolation of redundant sensors and channels.

4. Physical layout drawings illustrating separation of redundant indicating instruments.

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Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

5. Component and module quality and performance documentation, with particular emphasis on equipment used for post-accident monitoring.

6. Descriptions of the means for identifying redundant elements (such as cable, cable tray, component, module, and interconnecting wiring identifications).

7. Schematic and control panel display diagrams illustrating system level automatic bypass indication for deliberately bypassed safety-related components or systems.

Other ICSB areas of review associated with SRDI systems that are covered elsewhere are as follows:

1. Environmental design and qualification testing of electrical equipment are addressed in SRP Section 3.11.

2. Technical specification requirements imposed upon the operation of the SRDI are discussed in SRP Section 7.1.

The RSB identifies any changes or corrections to the listing of engineered safety features and reactor coolant system variables and components that require indication, by examining the tables of SAR Section 7.5 that describe the information display (including accuracy and range requirements of indicating instrumentation) required by the operator to perform manual safety functions. The CPB identifies any changes or corrections to the listing of reactor variables that require indication, and the ASB and CSB identify "balance-of-plant" variables and components that require indication and, where necessary, states the required locations of the indicators.

The PSB reviews the adequacy of physical separation criteria for cabling and electrical power supplies. It also determines that control power supplied to redundant indication channels is from appropriate redundant sources.

The MEB reviews, in SAR Section 3.10, the criteria for seismic qualification and the test and analysis procedures and methods to assure the operability of the SRDI.

The QAB reviews, in SAR Chapter 17, the quality assurance procedures to be used by the applicant in the design, construction, installation, and maintenance of the SRDI.

II. ACCEPTANCE CRITERIA

The safety-related display instrumentation design is acceptable when it can be concluded that it conforms to the criteria listed in Table 7-1 and that the operator will be provided with sufficient information to perform required manual safety functions should such action be necessary. Specific points with regard to these criteria are detailed below.

1. The SRDI should cover appropriate variables, consistent with the assumptions for accident analyses and with the information needs of the operators in normal, transient, and accident conditions. The design of the post-accident SRDI should conform to the recommendations of Regulatory Guide 1.97. The accuracy and range of indicating instrumentation should be consistent with the assumptions of the accident analyses. Any exceptions to these requirements will be referred to the appropriate branch for resolution on an individual case basis.

2. All monitoring channels should be redundant, to assure that wrong indication due to device malfunction will not cause false action or inaction on the part of the operator. Identification malfunctions can be identified by cross checking between redundant channels.

3. Redundant channels of safety-related display instrumentation should be isolated physically and electrically to assure that a single failure will not result in complete loss of information about a monitored variable. Single failures might include such possible faults as shorts or open circuits or interconnecting signal or power cables. It also includes single credible malfunctions or events that might cause a number of subsequent component, module, or channel failures. All SRDI should be capable of operating from onsite power. If signals from the post-accident monitoring equipment are used for control, the required isolation devices will be classified as part of the post-accident monitoring instrumentation. No credible failure at the output of an isolation device should prevent the associated monitoring channel from meeting minimum performance requirements considered in the design bases.

4. Capability should be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation. An acceptable way of accomplishing this would be by:
   a. Perturbating the monitored variable and observing the resulting indications.
   b. Introducing and varying a substitute input to the sensor of the same nature as the measured variable.
   c. Cross checking between channels that bear a known relationship to each other and that have readouts available.

   For channels which monitor a normally static parameter, provisions should be made to allow periodic testing in accordance with Regulatory Guide 1.22, thereby verifying channel operability.

5. An indication system should be provided covering bypassed or deliberately inoperable conditions of safety systems. Guidelines for the indication system are provided in Regulatory Guide 1.47 and Branch Technical Position ICSB 21.

6. Cables, cable trays, components, modules, and interconnecting wiring should be identified. The method used for identification and the scheme used to distinguish

between redundant cables, cable trays, components, modules, and interconnecting wiring are acceptable if they are in accordance with the recommendations of Regulatory Guide 1.75.

7. Components and modules should be of a quality consistent with the reliability requirements for safety-related systems. An acceptable quality would be that of components and modules that have been previously used in similar service conditions and have demonstrated low maintenance requirements and failure rates. Other means to demonstrate acceptable quality would be through analysis and testing of components and modules, in accordance with criteria cited in Table 7-1.

8. In order to assure that the requirements of General Design Criterion 1, "Quality Standards and Records," are met in the SRDI, the quality assurance program must satisfy the requirements of IEEE Std 336, as amplified by Regulatory Guide 1.30.

9. For those areas of review identified in subsection I as being the responsibility of other branches, the acceptance criteria are included in the applicable SRP sections for those branches.

III. REVIEW PROCEDURES

The objectives in the review of the SRDI are to determine that the plant display instrumentation is designed, constructed, and installed in accordance with the design criteria outlined in subsection II. In the CP review, the descriptive information, including the design bases and their relation to the criteria, preliminary analyses, piping and instrumentation diagrams (P&IDs), functional control diagrams, preliminary electrical diagrams, and preliminary physical arrangement drawings are examined to determine that there is reasonable assurance that the final implementation will meet all criteria. At the OL stage, the objectives are verified by review of the tables of variables and components to be monitored, indicated, and recorded; functional control diagrams, P&IDs, and electrical distribution diagrams; physical layout drawings; component and module quality considerations; the identification scheme for redundant systems; and the procedures for maintenance and checking of the availability of each system.

In certain instances, it will be the reviewer's judgment that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a nonuniform placement of emphasis are the introduction of new design features or the utilization of design features previously reviewed and found acceptable.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The review steps are as follows:

1. Based upon information provided by the RSB, CPB, CSB and ASB with regard to variables that need to be monitored, and upon Branch Technical Position ICSB 21, the list of monitored variables (if available) in the SAR is checked for sufficiency. In addition, the accuracy and range of the monitors are checked against the plant accident analyses as noted in II.1, above.

2. Functional control diagrams and P&IDs are reviewed to establish that the redundancy is sufficient, so that false indication due to malfunction of an indicating device should not lead to an undesirable manual action. In reviewing the P&IDs, the reviewer verifies that redundant sensors for each monitored variable are identified. After establishing sensor redundancy, the functional control diagrams are reviewed to ascertain that redundancy is maintained through the system logic down to the indicating devices.

3. Since independence from offsite power is required for post-accident SRDI, emphasis is placed on the electrical distribution system supplying power to post-accident SRDI. The PSB reviews electrical distribution diagrams to establish that redundant instrument channels are supplied from redundant electrical distribution channels of the emergency power supply. The ICSB reviews electrical schematic diagrams (as appropriate) to ascertain that there is no interconnecting wiring between redundant channels whose failure (open or short circuit) could cause the simultaneous loss of redundant channels. Also, the schematic diagrams are reviewed to ascertain that devices that isolate signals used for both safety indication and control are properly identified as part of the safety system and that a failure at the output of the isolation device does not prevent the associated monitoring channel from performing its safety function. Qualification of isolation devices is covered in the review of Sections 3.10 and 3.11 of the SAR.

4. Physical layout drawings (such as control room panel layouts, local panel layouts, sensor locations, instrument cabinet layout drawings, penetration drawings, and cable routing drawings) are reviewed to establish that physical independence is maintained between redundant channels of the SRDI. The control room panel layout drawings are examined to determine that the minimum separation distance between redundant equipment and circuits internal to the control boards is in accordance with Section 5.6 of Regulatory Guide 1.75. Local panel layout drawings are examined on the same basis. Sensor location drawings are examined to determine that the connections to the process system are sufficiently separated, in accordance with Section 5.8 of Regulatory Guide 1.75, to assure functional capability despite any single design basis event. The separation recommendations of Section 5.6 of this guide also apply to instrument cabinets (the layout drawings are examined to determine that the minimum separation distance between redundant equipment and circuits internal to the cabinets is provided). The procedure for review of penetration drawings and cable routing drawings is discussed in SRP Section 8.3.

5. With regard to the quality of components, there are at present no specific criteria to judge the quality of equipment used in the SRDI. However, Appendix B to 10 CFR Part 50 provides some guidance from which a judgment may be made of the quality of equipment required for the SRDI.

6. The procedure for reviewing the identification scheme proposed by the applicant to distinguish between redundant reactor protection system elements (including SRDI) is described in SRP Section 7.3.

7. The applicant's final design and installation of the SRDI is examined (schematic diagrams, wiring diagrams, installation drawings, etc.) to determine that the system includes the capability of periodic tests or checks to assure availability during operation.

IV. EVALUATION FINDINGS

The reviewer confirms that sufficient information has been provided and the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The safety-related display instrumentation provides the operator with information on the status of the plant to allow manual safety actions to be performed whenever necessary. The scope of review of safety-related display instrumentation included tables of system variables and component states to be indicated, functional control diagrams (CP and OL), electrical and physical layout drawings (OL), and descriptive information. The review has included the applicant's proposed design criteria and design bases, including that for indication of bypassed or inoperable safety-related systems. The review also has included the applicant's analyses of the manner in which the design of safety-related display instrumentation conforms to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for safety-related display instrumentation to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

"The staff concludes that the design of safety-related display instrumentation for the _____ plant conforms to applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (ICSB)."

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SECTION 7.6 ALL OTHER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Core Performance Branch (CPB)
Reactor Systems Branch (RSB)
Containment Systems Branch (CSB)
Auxiliary Systems Branch (ASB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The group of instrumentation systems reviewed under this SRP section are those required for safety that are not identified as part of the reactor protection system, engineered safety features systems, safety-related display instrumentation systems, or systems required for safe shutdown. They consist of fire protection and detection systems and groups of interlocks intended to protect other vital systems from potentially damaging transients during normal operation and under accident conditions. Examples of such interlocks are cold water interlocks, refueling interlocks, interlocks that prevent overpressurization of low pressure systems, reactor vessel instrumentation, and accumulator valve interlocks. They also include the process and effluent radiological monitors which should be reviewed for the adequacy of their seismic design, environmental design, redundancy and emergency power (See SRP Section 11.5).

The review of these systems encompasses the sensors, initiating circuits, logic elements, bypasses, interlocks, redundancy and diversity features, actuated devices, testing provisions, power supply, separation and equipment qualifications.

The ICSB has primary responsibility for the review of these systems with the exception of fire protection systems for which the ASB has primary responsibility. The review should confirm that these systems and essential supporting systems will perform design functions when required during all applicable operational and emergency conditions of the plant, and that the design of these systems conforms to all applicable acceptance criteria.

The descriptive information contained in the applicant's safety analysis report (SAR), including single line diagrams, electrical and instrumentation schematics, piping and instrumentation diagrams (P&IDs), and physical arrangement diagrams, is reviewed to

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Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

ascertain that "other instrumentation systems required for safety" meet the acceptance criteria discussed in SRP Section 7.1 and listed in Table 7-1. For a construction permit (CP) review, a commitment to meet these criteria may suffice in cases where the design of these systems has not been completed. For an operating license (OL) review, however, the actual design must be found to meet these criteria.

The primary review responsibility of the ICSB shall be to perform its reviews as assigned below, to coordinate and report the reviews of the portions reviewed by other branches as identified below.

1. The ICSB determines that adequate redundancy of logic and instrumentation has been provided for the operation and status monitoring of "other instrumentation systems required for safety." The reviews establishes that these systems can perform their necessary functions after sustaining a single failure. This requires the review of descriptive information contained in the SAR, functional diagrams, instrumentation schematics and P&IDs.

2. The PSB reviews the adequacy of physical separation criteria for the cabling and electrical power equipment, determines that control and motive power supplied to redundant systems is from the appropriate redundant power source and that the single failure criterion has been met in the design of manually-controlled electrically-operated valves.

3. The ICSB review assures that the instrumentation and control equipment, cabling and structures housing instrumentation portions of "other instrumentation systems required for safety" are designed in accordance with criteria required for Class 1E and seismic Category I systems and structures.

4. The PSB determines that electrical equipment, cabling, related cable tray systems and structures containing electrical portions of "other instrumentation systems required for safety" are designed in accordance with criteria required for Class 1E and seismic Category I systems and structures.

5. The ICSB and PSB, for their respective areas, verifies that all instrumentation and electrical equipment of "other instrumentation systems required for safety" have been included in the seismic qualification program.

6. The ICSB review assures that the instrumentation and control equipment and cabling is qualified by tests or analyses or by a combination of tests and analysis to perform its safety-related functions in environments that may develop as a result of design basis accidents or anticipated operational occurrences. The PSB review assures that similiar qualification is accomplished for the power and electrical equipment, cabling and cable tray systems.

7. The ICSB determines that on-line testability of the systems and indication of bypassed or inoperable status of these systems required for safety are provided for in the design.

It is the primary responsibility of the ASB to review the acceptability of the fire protection systems. Within this primary responsibility the ICSB reviews the fire detection systems and the initiation and control systems of the fire protection systems. The PSB similarly reviews the adequacy of the power supply for these systems and the required physical separation and isolation.

The ASB evaluates the adequacy of those auxiliary systems required for the proper operation of "other instrumentation systems required for safety." These include compressed air systems, air conditioning systems, heat tracing systems, etc. In addition, the ASB reviews the physical arrangement of components and structures related to "other instrumentation systems required for safety" and supporting systems, and determines that single events will not disable redundant parts of these systems. The CPB verifies that boron dilution rates achievable, or the accidental startup of an unborated or cold reactor coolant loop, result in acceptable reactivity insertion rates as discussed in SRP Section 4.3.

The CSB reviews the containment ventilation and atmosphere control systems provided to maintain environmental conditions required for operation of electrical and instrumentation equipment associated with "other instrumentation systems required for safety" and located inside containment.

The MEB review confirms that the seismic qualification of the instrumentation and electrical systems is acceptable. This includes the seismic design criteria, analyses, testing procedures, and restraint measures employed in the seismic design and installation of Category I instrumentation and electrical equipment including trays, control room boards, and instrument racks and panels, as covered in SRP Section 3.10.

The RSB review identifies "other instrumentation systems required for safety" and confirms that the configuration and design bases of the systems are correct, and that design parameters such as temperature, pressure, flow rate, and reactivity can be controlled within acceptable limits. Information is provided to the ICSB as to any corrections needed in the SAR and any exceptions to acceptance criteria taken by the applicant.

The QAB review verifies that the quality assurance program proposed by the applicant includes "other instrumentation systems required for safety."

II. ACCEPTANCE CRITERIA

The design, materials, qualification testing, and surveillance of "other instrumentation systems required for safety" are covered by several General Design Criteria (GDC), IEEE standards, regulatory guides, and branch technical positions which are applicable in whole or in part. A list of the applicable criteria, standards, guides, and branch positions is given in Table 7-1 and Appendix 7-A to this chapter.

The "other instrumentation systems required for safety" are acceptable when it is determined that these systems satisfy the following requirements:

1. They have the required redundancy.

2. They meet the single failure criterion.

3. They have the required capacity and reliability to perform intended safety functions on demand.

4. They are capable of functioning during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.

5. They are testable during reactor operation.

The criteria listed in Table 7-1 are utilized as the bases for determining that these requirements are met and that the "other instrumentation systems required for safety" are acceptable. How these criteria are applied during the review process is discussed in subsection III. Specific points with regard to the acceptance criteria are detailed below.

1. System Redundancy Requirements

   GDC 26 and 33 and IEEE Std 279 specify the requirements that "other instrumentation systems required for safety," among others, must meet with regard to all operating conditions (such as loss of offsite power), so that they can perform needed safety functions assuming a single failure. If a determination is made that these systems meet the requirements of these criteria, they are acceptable with regard to redundancy requirements.

2. Conformance with the Single Failure Criterion

   IEEE Std 279, IEEE Std 379, and Regulatory Guide 1.53 provide that safety systems should be capable of performing needed safety functions after sustaining a single failure. Regarding the application of the single failure criterion to the design of manually-controlled electrically-operated valves in safety systems, the acceptability of proposed designs is based on Branch Technical Position ICSB 18 (PSB). This position states that it is acceptable to disconnect electric power to a safety-related valve as a means of designing against an active valve malfunction. The requirements for tolerance of single failures in fire detection systems are given in NFPA 72D.

3. Identification of Cables and Raceways

   The method used for identifying power and signal cables and raceways as safety-related equipment, and the identification scheme used to distinguish between redundant cables, raceways, and instrument panels should be in accordance with the recommendations of Regulatory Guide 1.75.

4. Vital Supporting Systems

The instrumentation, control, and electric equipment associated with auxiliary systems that support "other systems required for safety" should meet the same acceptance criteria as the systems they support.

5. Testing, Quality Assurance, and System Availability Surveillance

GDC 1 and 21, IEEE Stds 279, 336, and 338; and Regulatory Guides 1.22, 1.47, 1.68, and 1.118 contain the applicable acceptance criteria with regard to preoperational and periodic testing, quality assurance, and design provisions for indicating the availability of "other instrumentation systems required for safety."

6. Fire Protection System

The staff has identified the National Fire Protection Codes (NFPA) as a generally recognized code that provides guidance for use in evaluating the fire detection systems and the supervisory systems for the fire suppression systems.

The following additional requirements and criteria apply specifically to the review by the ICSB of the instrumentation and control portions of the fire protection system.

1. All plants for which a construction permit application was docketed on or after July 1, 1976, must satisfy the requirements of NFPA 72D for Class A, Class 1 systems and have fire detection and suppression instrumentation and controls which can be manually connected to a Class 1E power source or automatically connected to a non-Class 1E onsite power sources which satisfies the requirements of NFPA 72D Section 2260.

2. All plants for which a construction permit application was docketed prior to July 1, 1976, must satisfy either position 1 above or the following minimum requirements.

   a. Fire detection and suppression systems instrumentation and controls must be supplied from a Class 1E power source if the requirements of NFPA 72D Sections 2222, 2223, and 2224 are not met. If NFPA Sections 2222, 2223, and 2224 are met the fire detection and suppression systems instrumentation and controls must have the capability of being manually connected to a Class 1E source.

   b. The wiring of the fire detection circuits must conform to the requirements of a Class 1 circuit as defined in NFPA 70. Where voltages exceed 30 volts, energy limiting devices must satisfy the requirements of Regulatory Guide 1.75 for isolation devices or a backup trip device shall be provided with demonstrated fully selective tripping. The purpose of this requirement is to allow older plants to use already-purchased non-NFPA 72D equipment provided that:

      (i) Equipment will continue to function during a loss of offsite power.
      (ii) A fault within the equipment will not disable the onsite power source.

(iii) A fault in one fire detection zone (which is assumed to be associated with a division) is not propagated to a second zone (which is assumed to be in a different division).

(iv) Actual tests of the installed equipment shall be conducted to demonstrate the capability of the backup fault protection equipment when fuses are used as primary protection, and actual tests of the primary protection shall be conducted when fuses are not used for primary protection.

c. Unless redundant, electrically isolated and physically separated systems which meet the requirements of Regulatory Guide 1.75 are provided, the detector wiring shall be sufficiently larger than the NFPA-72D minimum sizes, which are:

(i) 22 AWG stranded, 5 or more strands/conductor

(ii) 19 AWG stranded, 2 or more strands/conductor

(iii) 16 AWG single conductor solid

The goal of this requirement is to reduce incipient failures which may result from damage due to tensile stress and would be undetected between systems tests in an unsupervised system.

d. In order to reduce the possibility of undetected ground faults in unsupervised systems, fire detection circuits shall be run in conduit unless redundant, electrically isolated and physically separated systems which meet the requirements of Regulatory Guide 1.75 are provided.

e. Circuits which do not meet the supervisory requirements of NFPA 72D shall be tested no less frequently than monthly by causing the detector at the end of each branch circuit to actuate an alarm. Non-supervised heat detection cable tests shall include resistance measurements of each lead-to-ground and lead-to-lead taken at its equipment cabinet with one end of the loop opened. The design should include this capability without requiring the lifting of leads or the use of jumpers.

f. The other aspects of testing and system availability surveillance shall be in accordance with the requirements of NFPA 72D and included in the plant's Technical Specifications.

g. Fire suppression systems which are based on the use of suffocating gases shall sound a pre-discharge alarm in the hazard zone prior to release. The time delay before release shall be based on the actual design of the affected space and shall permit personnel to escape from the most remote point in the space to the most remote exit prior to gas release.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The review is conducted to ascertain that the designs of "other instrumentation systems required for safety" (or design commitments in the case of CP's) are acceptable in terms of the acceptance criteria listed in subsection II. The main objectives of the review of these systems are to determine that they include the required redundancy, meet the single failure criterion, provide the required capacity and reliability to perform intended safety functions on demand, and can function during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.

For a CP application, the descriptive information contained in the preliminary safety analysis report (PSAR), including the design bases and their justification with regard to the acceptance criteria, accident analyses, electrical single line and P&ID's, are reviewed to determine that the basic design features and the commitments made at this stage provide assurance that the final design will meet the acceptance criteria. During the OL review, it is verified that the acceptance criteria are met through review of the final electrical and instrumentation drawings and the physical layout drawings, and a site visit during which a spot-check verification of the design is performed.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The various elements of the review are carried out as follows:

1. The logic diagrams, electrical one-line diagrams, P&IDs (for CP and OL reviews), instrumentation and electrical schematics (for the OL review) as described in the SAR, are reviewed to verify that the necessary redundancy is provided. This review includes instrumentation channels used to sense vital parameters such as temperature, pressure, water level, etc., the associated logic and actuated devices, and the motive and control sources.

2. Conformance with the single failure criterion as specified by IEEE Std 279, IEEE Std 379, and Regulatory Guide 1.53 is verified by review of the same information as for redundancy and may be done, to some degree by necessity, at the same time. The guidance provided by Regulatory Guide 1.53 is excellent for ascertaining that a given design is single failure proof. A particularly important point to check is one cited in Position 4 of Regulatory Guide 1.53, where a single d-c source supplies control power for one channel of system logic and for the redundant actuator circuit.

3. For a multi-unit design where electrical systems are shared, resulting in numerous and more complex interaction modes, a fault-tree and decision-tree analysis may be requested from the applicant to show that single failures, or single events resulting in multiple failures, will not result in unacceptable consequences with respect to the capability of "other instrumentation systems required for safety" to perform safety functions when required. Additional guidance with regard to the single failure criterion is given in SRP Sections 7.2 and 7.3.

4. For manually-controlled electrically-operated valves in safety-related systems, the acceptability of proposed designs is based on Branch Technical Position ICSB 18 (PSB). This position basically states that it is acceptable to disconnect electric power to a safety-related valve as a means of removing the possibility of an active failure of that valve. The adequacy of the means for accomplishing such disconnection of power is reviewed by the PSB.

5. The PSB has the primary responsibility for reviewing the adequacy of physical separation, electrical isolation and identification of cables, raceways and equipment of redundant systems. Regulatory Guide 1.75 and Sections 5.1.2 and 5.6.3 of IEEE Std 384 provide guidance for satisfying the acceptance criteria with respect to the identification of power and signal cables, raceways, and instrument panels related to "other instrumentation systems required for safety." The criteria for identification and separation of redundant systems as discussed in IEEE Std 384 and Regulatory Guide 1.75 are presented in sufficient detail to make their application self-explanatory. GDC 1 and 21; and IEEE Stds 279, 336 and 338 provide the requirements that the design of these systems must meet with regard to preoperational and periodic testing supplemented by the guidance provided in Regulatory Guides 1.22, 147, 1.68, and 1.118. The primary review responsibility for preoperational testing is with the QAB. Periodic and downtime restrictions are specified in the technical specifications. The review procedures for technical specifications are covered in SRP Section 7.1.

6. The process of aligning various fluid or gaseous systems for certain modes of operation may involve the interconnection of high pressure and low pressure systems. During normal operation, these systems must be isolated from one another. For example, the residual heat removal (RHR) system of some reactor designs is interfaced with the high pressure reactor coolant system. There should be two isolation valves in series, with diverse interlocks that will prevent operation of these valves unless the primary reactor coolant pressure is below a predetermined value. For a detailed description of the isolation requirements, see Branch Technical Position ICSB 3.

7. The main steam line radiation monitoring system in boiling water reactors is provided to monitor the gross release of fission products in the reactor coolant and to initiate protective action if the level of such release exceeds a predetermined level. The ICSB reviewer should assure that the instrumentation channels provided for this purpose are divided into two redundant and independent groups. Also, the PSB should determine that the two groups are powered from independent busses of the Class 1E emergency power system.

   Normally, four gamma-sensitive channels are provided to monitor the radiation level in the main steam lines. The ICSB reviewer should assure that the geometric arrangement and physical location of these is such that a fission product release will be detected with any number of main steam lines in operation, and that it will be detected at the earliest possible time following a fuel failure. It is important

that the failure of any one of these four channels will not result in an inadvertent action. The initiating logic should be checked to make sure that this is the case. The ICSB reviewer should verify that the design has provisions for testing and that operability can be adequately tested.

8. The ICSB reviewer should verify that the "other instrumentation systems required for safety" have been qualified to operate under normal, operational transient, accident, and post-accident environmental conditions and that they satisfy the recommendations of IEEE Std 323. The ICSB also coordinates the verification that equipment and structures related to these systems are seismically qualified or designed, and that the seismic qualification and analysis program submitted by the applicant is acceptable to the MEB and PSB. The environmental qualification of components and cabling of these systems should be the same as for the systems discussed in SRP Sections 7.3 and 3.11.

9. An important part of the review is the engineering drawing review. A drawing review should include the following:

   a. Verification that a complete set of drawings has been submitted that includes schematics, logic diagrams, P&IDs, and location layout drawings for these systems.

   b. Verification that the submitted drawings represent the actual system designs and layouts for the particular plant, and that those intended to be "typical" of a system are so identified.

   c. Verification that the design and layout meet the applicable criteria listed in subsection II.

10. A site visit and inspection, in accordance with the procedure described in Appendix 7-B, should be performed before the evaluation findings are written for OL reviews. A site inspection should include spot-check verifications that the design and layout criteria are actually implemented at the hardware assembly stage. Items to investigate during the visit are included in the typical agenda given in Appendix 7-B.

11. For the fire protection system the ICSB reviews the design of the fire detection instrumentation with the exception of detector sensitivity and detector location. Also reviewed are the designs of the initiation, control and instrumentation systems for the fire suppression systems. The ICSB also evaluates the consequences of the effects of a failure or failures of the fire protection system on safety-related instrumentation equipment so that spurious initiations of the fire suppression systems do not become a source of common mode failure. In this regard the effects of exposure to the fire fighting medium on the Class 1E instrumentation and control equipment shall be considered.

The PSB reviews the design of power supply equipment and systems, physical separation, electrical isolation and cabling (including identification) as part of its review of fire protection electrical systems.

In certain instances, it will be the reviewer's judgment that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

ICSB coordinates its own verification and the verifications of the other branches that sufficient information has been submitted and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The other instrumentation systems required for safety consist of safety-related instrumentation systems not identified as parts of the reactor protection system, engineered safety features systems, safety-related display instrumentation systems, or systems required for safe shutdown. They consist of fire detection, initiation and control for fire suppression and alarm systems, and groups of interlocks intended to protect other vital systems from potentially damaging transients during normal operating, startup, shutdown, and accident conditions.

The review encompassed sensors, initiating units, logic, bypasses, interlocks, redundancy and diversity features, actuated devices, testing provisions, and equipment qualifications. The review included single line diagrams (CP & OL), schematic diagrams (OL), and descriptive information on this group of systems and supporting auxiliaries that are essential for their operation. The review also included the applicant's proposed design criteria and design bases, and analyses of the manner in which the design of these systems conforms to the proposed design criteria.

The basis for acceptance by the staff is conformance of the applicant's designs, design criteria, and design bases to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

The staff concludes that the design of these systems conforms to applicable regulations, guides, technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems.

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (ICSB)."

3. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

4. National Fire Protection Code, Section 72D; National Fire Protection Association, 470 Atlantic Ave., Boston, MA.

5. National Fire Protection Code, Section 70 (National Electric Code); ibid.

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SECTION 7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Auxiliary Systems Branch (ASB)
Reactor Systems Branch (RSB)
Quality Assurance Branch (QAB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The areas reviewed in this section of the applicant's safety analysis report (SAR) include those control systems identified by the applicant as being nonsafety-related. These control systems may include the primary system pressure, temperature and water level controls, and feedwater controls. The intent of the review is to assure that failures of these controls would not impair the protection system capability in any significant manner. Since the control systems of interest under this SRP section may vary from plant to plant depending upon individual designs, the applicant should identify all such systems and provide analyses to support their classification as nonsafety-related control systems.

The ICSB will review the following aspects of the nonsafety-related control systems: the circuit-to-circuit failure modes of a single nonsafety-related control system and their effect on the protection system, and gross failure modes of nonsafety-related control systems and their functional effect on the protection system.

The ASB, PSB and RSB provide assistance in verifying that all control systems have been identified and that the input signal parameters for the control systems are correct. The RSB determines that the control systems identified in this SAR section are not required for safety and that no credit is taken in the plant accident analyses for the control systems so identified.

The PSB reviews the turbine generator control and overspeed protection systems as set forth in SRP Section 10.2.

The QAB verifies that the quality assurance program implemented for control system components, where necessary, is adequate.

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Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

II. ACCEPTANCE CRITERIA

The control systems not required for safety are acceptable if failures of control system components or total systems would not significantly affect the ability of plant safety systems to function as required, or cause plant conditions more severe than those for which the plant safety systems are designed.

Table 7-1 lists those General Design Criteria (GDC) of Appendix A to 10 CFR Part 50, and standards of the Institute of Electrical and Electronic Engineers (IEEE), that are used as references in arriving at this conclusion. General Design Criteria 13 and 24 and IEEE Std 279, Section 4.7, are of special importance among these references.

1. Conformance with General Design Criterion 13 for Instrumentation and Control Requirements.
   Instrumentation should be provided to monitor variables and systems over their anticipated ranges for normal operation and for anticipated operational occurrences as appropriate to minimize challenges to safety systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

2. Conformance with General Design Criteria 24 for Separation of Control Systems from Protection Systems.
   The protection system shall be separated from control systems to the extent that failure of any single control system component or channel which is common to control and protection systems shall not violate the reliability, redundancy, and independence requirements of the protection system. The interconnections between the protection and control system shall be limited so as to assure that safety is not significantly impaired.

3. Conformance to IEEE Std 279, Section 4.7, for Control and Protection System Interaction.
   The direct circuit-to-circuit and functional interactions between control and protection systems for single random or multiple failures in the control system shall not prevent the protection system channel from meeting the minimum performance requirements specified in the design bases.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

1. The objectives in the review are:

   a. To establish that control systems identified as being nonsafety-related, which, depending upon plant design, may include the primary system pressure, temperature, and feedwater controls, and steam generator water level controls, are, in fact, not required for plant safety.

b. To verify that no credit is taken for the operability of these control systems in the plant accident analyses in Chapter 15 of the SAR.

c. To assure that failures of these control systems would not impair the capability of the protection system in any significant manner or cause plant conditions more severe than those for which the plant safety systems are designed.

d. To establish that control system designs meet applicable requirements of the General Design Criteria and industry standards with regard to independence between control and protection functions.

2. In the construction permit (CP) review the descriptive information, including the design bases and preliminary analyses, are reviewed to determine that there is reasonable assurance that the final design will meet these objectives. The RSB, PSB and ASB identify the plant systems whose control system designs are to be reviewed to verify that no credit is taken for their operability in the plant accident analyses. ICSB reviews the descriptive information provided for those systems at the construction permit stage to assure that control and protective functions are adequately separated, to assess the effects of control system failures, and to verify that commitments are made that such failures will be included in the plant safety design bases.

3. At the operating license (OL) stage, the objectives described in item 1., above, are verified during the review of control system schematics. At the operating license stage, ICSB reviews electrical schematic drawings for these control systems as necessary to assure that adequate attention has been given to the separation of control and protective functions and to possible effects of failures of these systems. The review includes interactions between control systems and effects on plant safety systems due to control system malfunctions or failures.

4. Upon request for the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

5. A typical review procedure for pressurized water reactor (PWR) primary and secondary control system functions follows:

a. The primary system pressure is maintained within specified limits by the use of pressurizer heaters and spray valves. The primary pressure control system description and schematics are reviewed:

(1) To confirm that the system will maintain the primary coolant pressures within prescribed limits for normal and transient operating conditions.
(2) To determine the effects of loss of power to the pressurizer heaters and spray valves.
(3) To determine the effects of loss of air to any pneumatically-operated valves in the spray system.

Assistance as needed is obtained from the RSB in evaluating these items.

b. To meet the requirements of General Design Criterion 24 and Section 4.7 of IEEE Std 279 on control system interactions with the protection system, loss of primary pressure control function is analyzed. Assistance is obtained from RSB in establishing the sequence of events that would follow. The evaluation should show that failure of the primary pressure control system would not significantly degrade the capability of the protection system. Also, the reviewer determines that where a random failure in the pressure control system can result in a plant condition requiring protective action and can also prevent proper action of a protection channel designed to protect against the condition, the remaining redundant channels will provide the protective action even when degraded by another random failure.

c. The system description and control schematics of the feedwater regulating system are reviewed to idenfity failure modes of the system components. Assistance is obtained from the RSB, PSB and ASB in identifying the control function parameters. The system actions are established for loss of air to the feedwater control valves and malfunction in the feedwater heater bypass valves. The reviewer should verify that manual override of the automatic control is designed into the system.

d. The reviewer evaluates the effects of multiple failures in control systems resulting from single events. Failures in the secondary system water level (i.e., feedwater flow and steam generator water level) controls are analyzed along with failure in the primary coolant pressure control, where a single event can cause these multiple failures. With the assistance from the RSB and ASB, the reviewer determines that control function failures of both primary pressure and secondary water level controls would not prevent the minimum required number of reactor protection system channels from tripping the reactor.

In certain instances, it will be the reviewer's judgment that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a nonuniform placement of emphasis are the introduction of new design features or the utilization of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

At the construction permit stage, it should be established that the information and commitments documented in the preliminary safety analysis report (PSAR) provide reasonable assurance that the final designs of nonsafety-related control systems will conform with the intent of this SRP section.

At the operating license stage, sufficient design detail for these control systems is reviewed to determine adequate conformance. Exceptions to the acceptance basis given in subsection II are identified, with a statement as to how these exceptions provide a conservative basis for engineering design of the affected control systems.

The reviewer verifies that sufficient information has been submitted and the review supports conclusions of the following type, to be included in the staff's evaluation report:

> "The staff has reviewed the controls for systems not required for safety, to determine the affects of failures or malfunctions of these controls on the reactor protection system and other plant safety-related systems. The nonsafety-related control systems are (identify control systems so designated). We conclude that failures or malfunctions of these controls would not be expected to degrade the capabilities of plant safety systems in any significant degree, or to lead to plant conditions more severe than those for which the safety systems are designed."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."

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APPENDIX 7-A BRANCH TECHNICAL POSITIONS (ICSB)

The ICSB Branch Technical Positions (BTPs) represent guidelines intended to supplement the acceptance criteria established in Commission regulations and regulatory guides, and in applicable IEEE standards. The BTPs originate in technical problems or questions of interpretation that arise in the detailed reviews of plant designs. The staff must make a judgment in each such case, in order to complete its review of the particular application. Where the same technical problem or question of interpretation arises in several cases, the staff's judgment on the point at issue is formalized in a BTP. The BTP is primarily an instruction to staff reviewers that outlines an acceptable approach to the particular issue and ensures a uniform treatment of the issue by staff reviewers. The approaches taken in the BTPs, like the recommendations of regulatory guides, are not mandatory, but do provide defined, acceptable, and immediate solutions to some of the technical problems and questions of interpretation that arise in the review process. In some instances, regulatory guides may be developed from BTPs after a sufficient experience in their use has accumulated.

All ICSB BTPs applicable to the SRP sections in Chapter 7 have been collected in this Appendix for convenience. Other ICSB BTPs applicable to Chapter 8 are presented in SRP Appendix 8-A. When another branch or division is assigned review responsibility for a BTP, that branch or division identified parenthetically as part of the BTP designation.

BTP ICSB	Branch Technical Positions of the Instrumentation and Control Systems Branch
1. (DOR)	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors.
3.	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System.
4. (PSB)	Requirements of Motor-Operated Valves in the ECCS Accumulator Lines.
5.	Scram Breaker Test Requirements - Technical Specifications.
9.	Definition and Use of "Channel Calibration" - Technical Specifications.

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12. Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service.

13. Design Criteria for Auxiliary Feedwater Systems.

14. Spurious Withdrawals of Single Control Rods in Pressurized Water Reactor.

15. (PSB) Reactor Coolant Pump Breaker Qualification (attached to SRP Appendix 8-A).

16. Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors.

18. (PSB) Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves (attached to SRP Appendix 8-A).

19. Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems.

20. Design of Instrumentation and Controls Provided to Accomplish Change-over from Injection to Recirculation Mode.

21. Guidance for Application of Regulatory Guide 1.47.

22. Guidance for Application of Regulatory Guide 1.22.

25. Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole.

26. Requirements for Reactor Protection System Anticipatory Trips.

BRANCH TECHNICAL POSITION ICSB 1 (DOR)

BACKFITTING OF THE PROTECTION AND EMERGENCY POWER SYSTEMS OF NUCLEAR REACTORS

A. BACKGROUND

The acceptance criteria used by the staff in the evaluation of protection and emergency power systems undergo improvement from time to time. With each change it is necessary to determine whether previously approved designs should be modified (backfitted) to meet the revised criteria. Ths determination is made on the basis of whether a significant incremental increase in safety of the plant would be obtained that would justify the various difficulties of the change.

The actions which raise the question of possible backfitting are:

1. Application for a full-term operating license for plants now operating with a provisional operating license.

2. Evaluation of a significant plant modification proposed by the staff or the licensee.

3. Application for a full-term operating license for plants now operating under DOD 91-B exemptions.

B. BRANCH TECHNICAL POSITION

For cases falling in the categories 1-3 in (A) above, the following apply;

1. Instrumentation and electric equipment essential to safety which must function in an accident environment should be analyzed or tested to demonstrated this capability.

2. Protection circuits essential to safety should meet the single failure criterion of Section 4.2 of IEEE 279.

3. Where d-c power is required for safety, redundant d-c sources should be provided and the d-c circuits should meet the single failure criterion.

4. For reactor plants supplying electric power to electric utility grids, redundant sources of onsite a-c power should be provided and the a-c circuits should meet the single failure criterion. This aspect of the design of research and test reactors should be evaluated on an individual case basis.

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 3
ISOLATION OF LOW PRESSURE SYSTEMS FROM THE HIGH PRESSURE REACTOR COOLANT SYSTEM

A. BACKGROUND

During normal and emergency conditions, it is necessary to keep low pressure systems that are connected to the high pressure reactor coolant system properly isolated in order to avoid damage by overpressurization or the potential for loss of integrity of the low pressure system and possible radioactive releases. There have been a number of recommendations for accomplishing this aim. Until a more definitive guide is published, the criteria in Part B, below, provide an adequate and acceptable design solution for this concern.

B. BRANCH TECHNICAL POSITION

The following measures should be incorporated in designs of the interfaces between low pressure systems and the high pressure reactor coolant system:

1. At least two valves in series should be provided to isolate any subsystem whenever the primary system pressure is above the pressure rating of the subsystem.

2. For system interfaces where both valves are motor-operated, the valves should have independent and diverse interlocks to prevent both from opening unless the primary system pressure is below the subsystem design pressure. Also, the valve operators should receive a signal to close automatically whenever the primary system pressure exceeds the subsystem design pressure.

3. For those system interfaces where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent the valve from opening whenever the primary pressure is above the subsystem design pressure, and to close automatically whenever the primary system pressure exceeds the subsystem design pressure.

4. Suitable valve position indication should be provided in the control room for the interface valves.

5. For those interfaces where the subsystem is required for ECCS operation, the above recommendations need not be implemented. System interfaces of this type should be evaluated on an individual case basis.

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 4 (PSB)
REQUIREMENTS OF MOTOR-OPERATED VALVES IN THE ECCS ACCUMULATOR LINES

A. BACKGROUND

For many postulated loss-of-coolant accidents, the performance of the emergency core cooling system (ECCS) in pressurized water reactor plants depends upon proper functioning of the safety injection tanks (also referred to as "accumulators" or "flooding tanks" in some applications). In these plants, a motor-operated isolation valve (MOIV) and two check valves are provided in series between each safety injection tank and the reactor coolant (primary) system.

The MOIVs must be considered to be "operating bypasses" because, when closed, they prevent the safety injection tanks from performing the intended protective function. IEEE Std 279 has a requirement for "operating bypasses" which states that the bypasses of a protective function will be removed automatically whenever permissive conditions are not met. This Branch Technical Position provides specific guidance in meeting the intent of IEEE Std 279 for safety injection tank MOIVs.

It should be noted that BTP ICSB 18 (PSB), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," also applies to these isolation valves and should be used in conjunction with this position.

B. BRANCH TECHNICAL POSITION

The following features should be incorporated in the design of MOIV systems for safety injection tanks to meet the intent of IEEE Std 279:

1. Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications), or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator.

2. Visual indication in the control room of the open or closed status of the valve.

3. An audible and visual alarm, independent of item (2), above, that is actuated by a sensor on the valve when the valve is not in the fully-open position.

4. Utilization of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the technical specifications).

C. REFERENCES

1. Arkansas 1, Unit 1, Safety Evaluation Report, January 23, 1973.

2. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

3. BTP ICSB 18 (PSB), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves."

BRANCH TECHNICAL POSITION ICSB 5
SCRAM BREAKER TEST REQUIREMENTS - TECHNICAL SPECIFICATIONS

A. BACKGROUND

There have been some inconsistencies in the description of scram circuit test procedures in FSARs and technical specifications requirements. Some FSARs for plants with Westinghouse reactors describe the scram circuit test procedures and include a position for testing the scram breakers, but there are no provisions for doing so in the proposed technical specifications. It is the purpose of this Branch Technical Position to establish a uniform practice in this matter.

B. BRANCH TECHNICAL POSITION

The requirement that control rod drive trip breakers be tested monthly should be included in all plant technical specifications issued. For a model, refer to the Oconee Technical Specifications page 4.1-4, Table 4.1-1, item 2.

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 9
DEFINITION AND USE OF "CHANNEL CALIBRATION" - TECHNICAL SPECIFICATIONS

A. BACKGROUND

In several PWR technical specifications, the term "channel calibration" was used to describe a "daily adjusatment" for amplifier gain of the nuclear instrumentation power range channels. This adjustment was performed to maintain agreement between the indicated reactor nuclear power level and the reactor thermal power calculation. This adjustment is not considered by the staff to be a channel calibration. A calibration procedure performed on a monthly basis requires the following:

1. Performance of a functional test using a simulated signal to verify bistable action (protective trips including rod block trips and permissive interlocks) on a monthly basis.

2. Calibration of the upper and lower chambers of each flux channel for axial offset utilizing the in-core detectors on a calendar quarter basis.

3. Performance of a functional test using a simulated signal to verify positive and negative rate bistable action on a monthly basis.

Performance of a total system response time is required during each refueling outage.

B. BRANCH TECHNICAL POSITION

The "daily adjustment," which does not fulfill the intent or requirements of a calibration procedure, should remain as a daily requirement but be deleted from the "channel calibration" category in the technical specifications.

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 12
PROTECTION SYSTEM TRIP POINT CHANGES FOR OPERATION WITH REACTOR COOLANT PUMPS OUT OF SERVICE

A. BACKGROUND

For the past several years, including a time prior to the development of IEEE Std 279, the staff has required automatic adjustment to more restrictive settings of trips affecting reactor safety by means of circuits satisfying the single failure criterion. The basis for this requirement is that the function can be accomplished more reliably by automatic circuitry than by a human operator. This design practice, which has also been adopted independently by the national laboratories and by much of industry, served as the basis for paragraph 4.15, "Multiple Set Points," of IEEE Std 279.

More recently, all applicants have stated that their protection systems were designed to meet IEEE Std 279. Paragraph 4.15 of IEEE Std 279 specified that where a mode of reactor operation requires a more restrictive set point, the means for ensuring use of the more restrictive set point shall be positive and must meet the other requirements of IEEE Std 279. A number of designs have been proposed and accepted which reliably and simply satisfy this requirement. During the review of some applications, however, certain design deficiencies have been found. The purpose of this position is to provide additional guidance on the application of Section 4.15 of IEEE Std 279.

B. BRANCH TECHNICAL POSITION

1. If more restrictive safety trip points are required for operation with a reactor coolant pump out of service, and if operation with a reactor coolant pump out of service is of sufficient likelihod to be a planned mode of operation, the change to the more restrictive trip points should be accomplished automatically.

2. Plants with designs not in accordance with the above should have included in the plant technical specifications a requirement that the reactor be shut down prior to changing the set points manually.

C. REFERENCES

1. Millstone-3 Safety Evaluation Report, September 24, 1973.

2. Beaver Valley-2 Safety Evaluation Report, October 10, 1973.

3. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

BRANCH TECHNICAL POSITION ICSB 13
DESIGN CRITERIA FOR AUXILIARY FEEDWATER SYSTEMS

A. BACKGROUND

The function of the auxiliary feedwater system in pressurized water reactors is to provide an emergency source of feedwater supply to the steam generators. It is required to ensure safe shutdown in the event of a main turbine trip with loss of offsite power. The system is also started on a safety injection signal. Feedwater is pumped to each steam generator through normally open control valves. It was found that in some plant designs the auxiliary feedwater system did not meet the single failure criterion. It is the purpose of this Branch Technical Position to provide guidance and to establish uniform requirements for acceptable designs of auxiliary feedwater systems.

B. BRANCH TECHNICAL POSITION

The auxiliary feedwater system should be capable of satisfying the system functional requirements after a postulated break in the auxiliary feedwater piping inside containment together with a single electrical failure. The basis for the position is that an auxiliary feedwater piping break would result in tripping the unit and, in turn, might cause loss of offsite power. Standard staff assumptions for analyzing postulated accidents include the assumption of loss of offsite power if the affected unit generator is tripped by the accident. Such a circumstance would leave the plant without adequate means for removal of afterheat even though the reactor coolant pressure boundary was intact, an unacceptable result. Plant heat removal systems must, in any postulated piping break, be capable of removing afterheat to the ultimate heat sink assuming a single electrical (active) failure anywhere in the auxiliary feedwater system or in the onsite power system.

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 14
SPURIOUS WITHDRAWALS OF SINGLE CONTROL RODS IN PRESSURIZED WATER REACTORS

A. BACKGROUND

Recent operating experience with PWRs and subsequent reviews of PWR designs with regard to the requirements of General Design Criteria 20 and 25 have shown that single failures can cause inadvertent single rod withdrawals. The intent of this Branch Technical Position is to provide specific guidance toward an acceptable interpretation and application of GDC 20 and 25.

B. BRANCH TECHNICAL POSITION

Applicants have to demonstrate compliance with the requirements of GDC 20 to 25. For this purpose, it has to be shown by analysis that the consequences of uncontrolled or erroneous withdrawal of a single control rod under any possible conditions of reactor operation does not result in exceeding specified acceptable fuel design limits. If the results of this analysis show that the limits may be exceeded, the applicant must provide the results of failure modes and effects analyses to show that a single failure occurring in the control system, or an operator error, will not cause the uncontrolled or erroneous withdrawal of a single control rod. If the results of these analyses show that it is possible for uncontrolled or erroneous withdrawal of single control rods to occur, and the specified fuel design limits could be exceeded as a result, then the protection system must be designed to detect and terminate the resulting transient before the fuel design limits are exceeded.

C. REFERENCES

1. Surry 3 and 4 Safety Evaluation Report, March 26, 1974.

2. General Design Criteria, Appendix A, 10 CFR 50.

BRANCH TECHNICAL POSITION ICSB 16
CONTROL ELEMENT ASSEMBLY (CEA) INTERLOCKS IN COMBUSTION ENGINEERING REACTORS

A. BACKGROUND

Certain control element assembly interlocks provided in Combustion Engineering designs have not been treated as safety-related. It has been determined by the staff that, unless it can be shown by analysis that these interlocks are not required to assure fuel integrity, they should be treated as required for safety.

B. BRANCH TECHNICAL POSITION

The following interlocks in CE designs are considered safety-related, and unless it can be substantiated otherwise by supporting analyses, they should be designed to meet the requirements of IEEE Std 279. The interlocks in question are intended to prevent the following actions:

1. Insertion of shutdown CEAs before the regulating CEAs are inserted.

2. Simultaneous withdrawal of more than two groups of CEAs.

3. Withdrawal of a CEA group or groups out of proper sequence.

C. REFERENCE

1. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

BRANCH TECHNICAL POSITION ICSB 19
ACCEPTABILITY OF DESIGN CRITERIA FOR HYDROGEN MIXING AND DRYWELL VACUUM RELIEF SYSTEMS

A. BACKGROUND

Certain design problems arise from the containment design concept which utilizes a drywell and suppression pool for heat removal after a loss-of-coolant accident (LOCA). Two such problems are (1) the hydrogen concentration in the drywell may, in a relatively short time, exceed the limits described in BTP CSB 6-2 (a safety-related problem), and (2) eventual cooling of the drywell will cause steam to condense, resulting in a partial vacuum which can draw water from the suppression pool and partially flood the drywell (a problem related to equipment deterioration and repair costs, not safety).

A hydrogen mixing system is proposed to mix the atmosphere in the larger containment volume outside the drywell with that in the drywell, thereby reducing the overall hydrogen concentration to an acceptable level. In some designs, the hydrogen mixing system bypasses the suppression pool, resulting in an additional load on the containment heat removal system, and in the possibility of overpressurizing the containment. (There are times during a LOCA when bypassing the suppression pool would quickly overpressurize the containment.)

Some designs propose to avoid flooding of the drywell by means of a vacuum relief system utilizing the valves of the hydrogen mixing system.

In view of the stresses to which the reactor operator might be subject during and following a LOCA, it has been concluded that automatic as well as manual initiation at the system level should be provided in BWR 6/Mark III plants.

B. BRANCH TECHNICAL POSITION

1. The design of the hydrogen mixing system should provide for both manual and automatic initiation and should conform to all criteria for protection systems, including the provisions of IEEE Std 279 and Regulatory Guides 1.22 and 1.62. Automatic initiation should come from the sensors which sense that the hydrogen concentration in the drywell has exceeded the limits described in BTP CSB 6-2.

2. The design should provide interlocks in both the automatic and manual circuits that will preclude the opening of valves which bypass the suppression pool before blowdown is complete.

3. If the hydrogen mixing system bypasses the suppression pool, the containment heat removal system should be automatically initiated whenever the hydrogen mixing system is initiated.

4. The containment heat removal system should be automatically initiated upon indication of high pressure in the containment.

5. In conformance with paragraph 4.8 of IEEE Std 279, all signal inputs to the hydrogen mixing system and to those portions of the vacuum relief system which are common to the hydrogen mixing system, should be direct measures, to the extent practical, of the desired variable. Exceptions should be identified and justified.

C. REFERENCES

1. Branch Technical Position CSB 6-2, "Guidelines for the Evaluation of the Bypass Leakage in Dual Containment Plants," attached to Standard Review Plan 6.2.5.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. Regulatory Guide 1.62, "Manual Initiation of Protection Actions."

BRANCH TECHNICAL POSITION ICSB 20
DESIGN OF INSTRUMENTATION AND CONTROLS PROVIDED TO ACCOMPLISH CHANGEOVER FROM INJECTION TO RECIRCULATION MODE

A. BACKGROUND

Designs are reviewed with regard to the automatic and manual initiation of protective actions, as set forth in paragraph 4.17 of IEEE Std 279. For some recent designs, the staff concluded that the proposed design of the circuits used to change over to the recirculation mode of operation following a loss-of-coolant accident did not conform to IEEE Std 279, and the complexity of the proposed changeover procedure raised questions as to whether the operator could be expected to perform correctly the required actions within the time and based on the information available to him.

B. BRANCH TECHNICAL POSITION

1. A design that provides manual initiation at the system level of the transfer to the recirculation mode, while not ideal, is sufficient and satisfies the intent of IEEE Std 279 provided that adequate instrumentation and information display are available to the operator so that he can make the correct decision at the correct time. Furthermore, it should be shown that, in case of operator error, there are sufficient time and information available so that the operator can correct the error, and the consequences of such an error are acceptable.

2. Automatic transfer to the recirculation mode is preferable to manual transfer, for the reasons cited above, and should be provided for standard plant designs submitted for review on a generic basis under the Commission's standardization policy.

C. REFERENCES

1. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

BRANCH TECHNICAL POSITION ICSB 21
GUIDANCE FOR APPLICATION OF REGULATORY GUIDE 1.47

A. BACKGROUND

The recommendations of Regulatory Guide 1.47 need further detailing as to methods of providing an acceptable design for the bypass and inoperable status indicators for engineered safety feature (ESF) systems. The purpose of this Branch Technical Position is to provide supplemental guidance for implementation of the recommendations of Regulatory Guide 1.47.

B. BRANCH TECHNICAL POSITION

The design criteria for bypass and inoperable status indication systems for ESF should reflect the importance of providing accurate information for the operator and reducing the possibility for the indicating equipment to affect adversely the monitored safety systems. In developing the design criteria, the following should be considered:

1. The bypass indicators should be arranged to enable the operator to determine the status of each safety system and determine whether continued reactor operation is permissible.

2. When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.

3. Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design.

4. Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety. Administrative procedures should not require immediate operator action based soley on the bypass indications.

5. The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on plant safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems.

6. The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified.

C. REFERENCES

1. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

BRANCH TECHNICAL POSITION ICSB 22
GUIDANCE FOR APPLICATION OF REGULATORY GUIDE 1.22

A. BACKGROUND

A recent application listed eight functions that are not tested while the reactor is operating at power. The applicant claimed that the periodic testing complied with Regulatory Guide 1.22. Regulatory Guide 1.22 does make provisions for actuated equipment that is not tested during reactor operation but it does not have provisions for excluding any portion of the protection system from the requirements of paragraphs 4.9 and 4.10 of IEEE Std 279.

B. BRANCH TECHNICAL POSITION

All portions of the protection systems should be designed in accordance with IEEE Std 279, as required by 10 CFR §50.55a(h). All actuated equipment that is not tested during reactor operation should be identified and a discussion of how each conforms to the provisions of paragraph D.4 of Regulatory Guide 1.22 should be submitted.

C. REFERENCES

1. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

2. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

BRANCH TECHNICAL POSITION ICSB 25
GUIDANCE FOR THE INTERPRETATION OF GENERAL DESIGN CRITERION 37 FOR TESTING THE OPERABILITY OF THE EMERGENCY CORE COOLING SYSTEM AS A WHOLE

A. BACKGROUND

General Design Criterion 37 requires, in part, that the emergency core cooling system be designed to permit testing the operability of the system as a whole under conditions as close to design as practical. It is stated in one recent application that the safety injection and residual heat removal pumps are made inoperable during the system tests.

B. BRANCH TECHNICAL POSITION

In order to comply with the requirements of GDC 37, all ECCS pumps should be included in the system test.

C. REFERENCES

1. General Design Criteria, Appendix A, 10 CFR 50.

BRANCH TECHNICAL POSITION ICSB 26
REQUIREMENTS FOR REACTOR PROTECTION SYSTEM ANTICIPATORY TRIPS

A. BACKGROUND

Several reactor designs have incorporated a number of anticipatory or "back-up" trips for which no credit was taken in the accident analyses. These trips, as a rule, were not designed to the requirements of IEEE Std 279 and therefore introduced nonsafety grade equipment into the reactor protection system. It was determined by the staff that this was not an acceptable practice, because of possible degradation of the reactor protection system.

B. BRANCH TECHNICAL POSITION

All reactor trips incorporated in the reactor protection system should be designed to meet the requirements of IEEE Std 279, without exception. This position applies to the entire trip function from the sensor to the final actuated device.

C. REFERENCES

1. Shearon Harris Safety Evaluation Report, September 15, 1972.

2. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

APPENDIX 7-B GENERAL AGENDA, STATION SITE VISITS

An important part of the review at the operating license stage is a site visit. It is preferable to have the site visit sometime before the completion of the drawing review. The purpose of the site visit is to supplement the review of the design based on the drawings and to evaluate the actual implementation of the design as installed at the site. The Regional Office of Regulatory Operations having jurisdiction over the plant under consideration should be notified ahead of time of the visit so that the regional inspectors can become familiar on a first-hand basis with findings that may require followup action. Since proper implementation of design is the ultimate goal of the technical review process, the importance of a site visit is self-evident. The following is a typical general agenda that may be used as a guide for developing a specific agenda for the plant under review.

1. Preliminary Discussions
   a. Unresolved items.
   b. Plant layout for touring.
   c. Special interest areas.

2. Control Room
   a. General layout.
   b. Nuclear and reactor protection instrument arrangement and layout.
   c. Rod position indication.
   d. Protection system initiation and bypass switch arrangements.
   e. Diesel control board.
   f. Cabling in control room (separation, loading, etc.).
   g. Radiation monitoring.
   h. Engineered safety feature initiation and bypass switch arrangements and status panels.

3. Cable Runs and Cable Spreading Area
   a. General layout.
   b. Degree of separation.
   c. Diverse wiring.
   d. Tray or wireway density (percentage fill).

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

e. Fire detection and protection.
f. Penetrations and cable terminations.

4. Switchgear Rooms
   a. General layout.
   b. Physical and electrical separation of redundant units.
   c. Potential for damage due to fire, missiles, etc.
   d. Cable installation.
   e. Fire detection and protection.

5. Battery Installations
   a. General layout.
   b. Physical and electrical separation.
   c. Potential for damage due to fire, missiles, etc.
   d. Fire detection and protection and security.
   e. Ventilation independence.
   f. Monitoring instrumentation.

6. Diesel Generators
   a. General layout.
   b. Physical and electrical separation of redundant units.
   c. Fuel supply system.
   d. Fire detection and protection.
   e. Qualification tests - interlocks and control panel.
   f. Auxiliary systems - starting air, combustion air, ventilation.

7. Instrument Piping
   a. Physical separation and single failure.
   b. Potential for damage due to fire, flooding, etc.
   c. Test features.

8. Transformers (Switchyard)
   a. Physical and electrical separation.
   b. Potential for damage due to fire, flooding, missiles, etc.
   c. Fire detection and protection.

9. Quality Control
   a. Onsite receipt, storage, installation, and protection procedures of installed instrumentation, equipment, and cables.

10. Reactor Building and Turbine Building
   a. Protection system instrument arrangement and layout.
   b. Potential for instrument damage due to fire, missiles, etc.
   c. Separation of piping and wiring to redundant instruments.
   d. Provisions for testing protection instruments.

11. Shared Systems for Multi-Unit Sites
    a. Equipment location and potential for damage.
    b. Control room control and assignment to accident unit.
    c. Availability upon completion of first unit.

12. Steam Lines - Main, HPCI, RCIC
    a. BWR temperature and radiation monitoring systems.
    b. Isolation valves.

13. Recirculation Water System (Condenser)
    a. Break detection and flood protection features.

14. Shutdown Outside Control Room
    a. Location for potential damage.
    b. Feedwater system, etc.