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

SECTION 10.2 TURBINE GENERATOR

REVIEW RESPONSIBILITIES

Primary - Power Systems Branch (PSB)

Secondary - Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Radiological Assessment Branch (RAB)
Auxiliary Systems Branch (ASB)

I. AREAS OF REVIEW

Nuclear reactor plants include a turbine generator system (TGS) to convert the energy in steam from the nuclear steam supply system into electrical energy. The TGS consists essentially of the turbine unit and the automatic devices, alarms, and trips which control and regulate turbine action, and the generator unit and its controls. The turbine control system and the steam inlet stop and control valves, the low pressure turbine steam intercept and inlet control valves, and the extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant.

The turbine generator system installed in a nuclear plant is typically equipped with redundant overspeed protection instrumentation and controls and the main steam and reheat steam control and stop valving arrangements typically provide redundancy in the valves essential for overspeed protection. The intent of the review under this SRP section is to verify that such redundancy, in conjunction with inservice inspection and testing of the essential valves, makes a turbine overspeed condition above the design overspeed very unlikely. Assessment of the risk to essential plant systems and structures from potential turbine missiles is reviewed under SRP Section 3.5.1.3.

1. The PSB reviews the turbine generator system and the components and subsystems normally provided with this equipment with respect to the following considerations:

   a. The general arrangement of the turbine and associated equipment with respect to safety-related structures and systems and balance of plant.

   b. The types and locations of main steam stop and control valves, reheat stop and intercept valves, and associated piping arrangements.

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c. The capability of the turbine generator control and overspeed protection systems to detect a turbine overspeed condition and to actuate appropriate system valves or other protective devices to preclude an overspeed condition above the design overspeed.

d. The overspeed protection instrumentation and controls with respect to redundancy, testability and reliability.

2. The PSB reviews the inservice inspection and operability assurance program for valves essential for overspeed protection.

3. The PSB reviews the applicant's proposed technical specifications for operating license applications as they relate to areas covered in this SRP section.

Secondary reviews are performed by other branches and the results used by the PSB to complete the overall evaluation of the system. The secondary reviews are as follows: the ASB determines that appropriate seismic and quality group classifications have been established for system components where appropriate. The ASB and MEB determine that the TGS is in accordance with Branch Technical Positions ASB 3-1 and MEB 3-1 as related to pipe cracks or breaks in high and moderate energy piping systems outside of containment. The MEB confirms that the components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB verifies that inservice inspection requirements are met for system components, and will verify the compatibility of the materials of construction with service conditions. The RAB determines if any radiation shielding is necessary to assure safe access to turbine equipment.

II. ACCEPTANCE CRITERIA

There are no general design criteria or regulatory guides that are directly applicable to the design evaluation of the turbine generator. Acceptability of the design of the turbine generator system, as described in the applicant's Safety Analysis Report (SAR), is based on the specific criteria listed below and on the similarity of the design to that of plants previously reviewed and found acceptable.

1. A turbine control and overspeed protection system should be provided to control turbine action under all normal or abnormal operating conditions, and to assure that a full load turbine trip will not cause the turbine to overspeed beyond acceptable limits. Under these conditions, the control and protection system should permit an orderly reactor shutdown either by use of the turbine bypass system and main steam relief system or other engineered safety systems. The overspeed protection system should meet the single failure criterion and should be testable when the turbine is in operation.

2. Turbine main steam stop and control valves and reheat steam stop and intercept valves should be provided to protect the turbine from exceeding set speeds and to protect the reactor system from abnormal surges. The reheat stop and intercept

valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to assure that turbine overspeed is controlled within acceptable limits. The valve arrangements and valve closure times should be such that a failure of any single valve to close will not result in excessive turbine overspeed in the event of a TGS trip signal.

3. The extraction steam check valves provided at extraction connections shall be capable of closing within an appropriate time limit to maintain stable turbine speeds in the event of a TGS trip signal.

4. The TGS should be provided with the capability to permit periodic testing of components important to safety while the unit is operating at rated load.

5. The inservice inspection program for main steam and reheat valves should include the following provisions:

   a. At approximately 3-1/3-year intervals, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by Section XI of the ASME Code for reactor components, at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve should be dismantled and visual and surface examinations conducted of valve seats, disks, and stems. If unacceptable flaws or excessive corrosion are found in a valve, all other valves of that type should be dismantled and inspected. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance.

   b. Main steam stop and control valves and reheat stop and intercept valves should be exercised at least once a week by closing each valve and observing by the valve position indicator that it moves smoothly to a fully closed position. At least once a month, this examination should be made by direct observation of the valve motion.

6. Unlimited access to all levels of the turbine area under all operating conditions should be provided. Radiation shielding should be provided as necessary to permit access.

7. Connection joints between the low pressure turbine exhaust and the main condenser should be arranged to prevent adverse effects on any safety-related equipment in the turbine room in the event of rupture (it is preferable not to locate safety-related equipment in the turbine room).

8. Branch Technical Positions ASB 3-1 and MEB 3-1 should be used to determine the acceptability of the effects of postulated TGS piping failures on safety-related equipment.

9. Regulatory Guide 1.68 should be used with regard to preoperational and startup testing of the power conversion system.

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 and bases and preliminary design as set forth in the Preliminary Safety Analysis Report meet the acceptance criteria given in subsection II. For review of 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 review procedures for OL applications include 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 as a result of the staff's review.

The review procedures given are for a typical turbine generator system. Any variance of the review, to take account of a proposed unique design, will be such as to assure that the system meets the criteria of subsection II. The reviewer evaluates the TGS, subsystems, and components of the unit that are considered essential for the safe integrated operation of the reactor facility. 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 SAR is reviewed to determine that the system description and piping and instrumentation diagrams (P&IDs) show the turbine generator system. The general arrangement of the TGS and associated equipment with respect to safety-related structures, systems, and components is noted.

2. The reviewer verifies the adequacy of the control and overspeed protection system and determines that:

   a. Support systems, subsystems, control systems, and alarms and trips will function for all abnormal conditions, including a single failure of any component or subsystem, and will preclude an unsafe turbine overspeed. The indepth defense that is provided by the turbine generator protection system to preclude excessive overspeeds should be designed with diverse protection means.

b. For normal speed-load control, the speed governor action of the electro-hydraulic control system fully cuts off steam at approximately 103 percent of rated turbine speed by closing the control, stop, and intercept valves.

c. A mechanical overspeed trip device is provided that will actuate the control, stop, and intercept valves at approximately 111 percent of rated speed.

d. An independent and redundant backup electrical overspeed trip circuit is provided that senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 112 percent of rated speed. This backup electrical overspeed trip system may utilize the same sensing techniques as the electro-hydraulic control system. However, the circuitry is reviewed to determine that the control signals from the two systems are isolated from and independent of one another.

3. The main steam stop and control and the reheat stop and intercept valving arrangements and valve closure times are reviewed to ensure that no single valve failure can disable the overspeed control function.

4. The extraction steam valving arrangements and valve closure times are reviewed to see that stable turbine operation will result after a TGS trip.

5. The capability for testing of essential components during TGS operation is reviewed.

6. The proposed inservice inspection program for essential speed control valves is reviewed to verify that it includes the provisions of item 5 of subsection II.

7. The reviewer consults with RAB with regard to expected radiation levels around the TGS and the degree of access to TGS components during operation.

8. If there are safety-related systems or portions of systems located close to the TGS, the physical layout of the system is reviewed to assure that protection has been provided from the effects of high and moderate energy TGS piping failures or failure of the connections from the low pressure turbine section of the main condenser. The means of providing such protection will be given 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 turbine generator system includes all components and equipment normally provided including turbine main steam stop and control valves and reheat steam stop and intercept valves. The scope of review of the turbine generator system for the _____ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and for control and supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the turbine generator system and the requirements for safe operation of the system during normal, abnormal, and accident conditions (CP).] [The review has determined that the design of the turbine generator system and supporting systems is in conformance with the 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 turbine generator system and supporting systems to applicable staff technical positions and industry standards.

"The staff concludes that the design of the turbine generator system conforms to all applicable staff positions and industry standards, and is acceptable."

V. REFERENCES

1. 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.

2. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

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SECTION 10.2.3 TURBINE DISK INTEGRITY

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. 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 turbine disks have large masses and rotate at relatively high speeds during normal reactor operation, failure of a disk could result in high energy missiles and excessive vibration of the turbine rotor assembly.

The following areas of the applicant's safety analysis report (SAR) relating to turbine disk integrity are reviewed:

1. Materials Selection

The high pressure turbine rotor is generally a one piece forging. Stresses at normal operating speeds are relatively low. The low pressure turbine rotor assembly usually consists of a rotor shaft with shrunk-on disks. Low pressure disk stresses are due to thermal gradients, the interference fit, and centrifugal forces. These stresses are relatively high. The low pressure turbine also operates at lower temperatures than the high pressure turbine. Thus, it is particularly important that low pressure disks be made of a tough material. The use of suitable material, adequate design, and inservice inspection can greatly reduce the probability of a turbine rotor or disk failure.

The materials properties, including descriptions of the procedures to minimize flaws and improve fracture toughness, are reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the low pressure disk materials. The materials properties such as creep, stress rupture, and toughness are reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the high pressure turbine rotor materials.

Included in this information are:

a. A discussion of the ductile-brittle transition temperatures (FATT or NDT) of the materials and the tests and standards used to determine them.

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b. The Charpy V-notch test program used to establish minimum upper shelf energies of the disk materials.

c. The fracture toughness test program used to establish minimum upper shelf toughness of the disk materials.

2. Fracture Toughness

The fracture toughness of the materials and the materials tests or correlations of Charpy and tensile data to toughness properties are reviewed to establish that the rotor and disk materials exhibit adequate fracture toughness at normal operating temperature and during startup.

3. High Temperature Properties

The creep and stress-rupture properties of the high pressure rotor materials are reviewed to establish that these materials exhibit adequate high-temperature, long-term properties.

4. Preservice Inspection

The preservice inspection program information is reviewed to verify that the disk forgings are first machined with minimum excess stock prior to heat treatment, that visual and surface inspections are performed on all finished machined surfaces, and that a 100% volumetric (ultrasonic) examination is performed.

5. Turbine Disk Design

The high and low pressure turbine rotor design information, including allowable stresses, temperature distributions, and design overspeed considerations, is reviewed.

6. Inservice Inspection

Descriptions of the baseline and inservice phases of the inservice inspection program, including types of inspections, areas to be inspected, frequencies of inspection, and acceptance criteria, are reviewed.

II. ACCEPTANCE CRITERIA

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

1. Materials Selection

The applicant's selection of a disk or rotor material is acceptable if in accordance with the following:

The turbine disk or rotor should be made from a material and by a process that tends to minimize flaw occurrence and maximize fracture toughness properties, such as a NiCrMoV alloy processed by vacuum melting or vacuum degassing. The material should be examined and tested to meet the following criteria:

a. Chemical analysis should be made for each forging. Elements that have a deleterious effect on toughness, such as sulfur and phosphorus, should be controlled to low levels.

b. The fracture appearance transition temperature (50% FATT) as obtained from Charpy tests performed in accordance with specification ASTM A-370 should be no higher than 0°F for low pressure disks and 50°F for high pressure rotors. Nil-ductility transition (NDT) temperature obtained in accordance with specification ASTM E-208 may be used in lieu of FATT. NDT temperatures should be no higher than -30 and 20°F, respectively.

c. The Charpy V-notch (C_v) energy at the minimum operating temperature of each low pressure disk in the tangential direction should be at least 60 ft-lbs. The C_v energy of high pressure rotor materials at minimum operating temperature should be at least 50 ft-lbs. A minimum of three C_v specimens should be tested in accordance with specification ASTM A-370.

2. Fracture Toughness

The low pressure turbine disk and high pressure rotor fracture toughness properties are acceptable if in compliance with the following criteria:

The ratio of the fracture toughness (K_{Ic}) of the disk and rotor materials to the maximum tangential stress at speeds from normal to design overspeed should be at least two $\sqrt{in.}$, at minimum operating temperature. Bore stress calculations should include components due to centrifugal loads, interference fit, and thermal gradients. Sufficient warmup time should be specified in the turbine operating instructions to assure that toughness will be adequate to prevent brittle fracture during startup. Fracture toughness properties can be obtained by any of the following methods:

a. Testing of the actual material of the turbine disk to establish the K_{Ic} value at normal operating temperature.

b. Testing of the actual material of the turbine disk with an instrumented Charpy machine and a fatigue precracked specimen to establish the K_{Ic} (dynamic) value at normal operating temperature. If this method is used, K_{Ic} (dynamic) shall be used in lieu of K_{Ic} (static) in meeting the toughness criteria above.

c. Estimating of K_{Ic} values at various temperatures from conventional Charpy and tensile data on the disk material using methods presented by J. A. Begley and W. A. Logsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1 (Ref. 5). This method of obtaining K_{Ic} should be used only on materials which exhibit a well-defined Charpy energy and fracture appearance transition curve and are strain-rate insensitive. The test data and the calculated toughness curve should be submitted to the staff for review.

d. Estimating "lower bound" values of K_{Ic} at various temperatures using the equivalent energy concept of F. J. Witt and T. R. Mager, ORNL-TM-3894 (Ref. 6). Load-displacement data from the compact tension specimens and the calculated toughness data should be submitted to the staff for review.

3. High Temperature Properties

The stress-rupture properties of the high pressure rotor material are acceptable if they provide sufficient assurance of rotor integrity for the lifetime of the turbine. The applicant can demonstrate compliance by submitting stress-rupture data of an equivalent material with similar properties to the staff for evaluation.

4. Preservice Inspection

The applicant's preservice inspection program is acceptable if in compliance with the following criteria:

a. Disk forgings should be rough machined prior to heat treatment.

b. Each finished disk should be subjected to 100% volumetric (ultrasonic), surface, and visual examinations using procedures and acceptance criteria equivalent to those specified for Class 1 components in the ASME Boiler and Pressure Vessel Code, Sections III and V.

c. Finish machined bores, keyways, and drilled holes should be subjected to magnetic particle or liquid penetrant examination. No flaw indications in keyway or hole regions are allowable.

d. Each turbine rotor assembly should be spin tested at the maximum speed anticipated during a turbine trip following loss of full load.

5. Turbine Disk Design

The applicant's design is acceptable if in compliance with the following:

The turbine assembly should be designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. The design of the turbine assembly should meet the following criteria:

a. The design overspeed of the turbine should be 5% above the highest anticipated speed resulting from a loss of load. The basis for the assumed design overspeed should be submitted to the staff for review.

b. The combined stresses of low pressure disks or high pressure rotors at design overspeed due to centrifugal forces, interference fit, and thermal gradients should not exceed 0.75 of the minimum specified yield strength of the material, or 0.75 of the measured yield strength in the weak direction of the materials if appropriate tensile tests have been performed on the actual disk material.

c. The turbine shaft bearings should be able to withstand any combination of the normal operating loads, anticipated transients, and accidents resulting in turbine trip.

d. The natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20% overspeed should be controlled in the design and operation so as to cause no distress to the unit during operation.

e. The turbine rotor and disk design should facilitate inservice inspection of all high stress regions, including bores and keyways, with the need for removing the disks from the shaft.

6. Inservice Inspection

The applicant's inservice inspection program is acceptable if in compliance with the following criteria:

The inservice inspection program for the steam turbine assembly should provide assurance that disk flaws that might lead to brittle failure of a disk at speeds up to design speed will be detected. The inservice inspection program for the turbine assembly should include the following:

a. Disassembly of the turbine at approximately 10-year intervals, during plant shutdown coinciding with the inservice inspection schedule as required by ASME Boiler and Pressure Vessel Code, Section XI, and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, low pressure turbine blades, low pressure disks, and high pressure rotors. This inspection should consist of visual, surface, and volumetric examinations, as required.

The applicant should keep abreast of technological advances in volumetric examination techniques so that when improved methods for inspection of turbine disks are developed they can be incorporated into the inservice inspection program.

b. An in-place visual examination of the turbine assembly at accessible locations should be conducted during refueling shutdowns at intervals not exceeding three years.

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 procedures are followed:

1. Materials Selection

The materials properties and the procedures to minimize flaws and improve fracture toughness, as described by the applicant, are reviewed and compared with the requirements of Section II.1 of this plan. If a new material not used in prior licensed cases

is utilized, the applicant's materials selection is reviewed and evaluated to establish its acceptability. Such an evaluation is based on the acceptance criteria of Section II of this review plan.

2. Fracture Toughness

The fracture toughness properties of the low pressure disk and high pressure rotor materials, including materials specimen test data, where applicable, are reviewed and compared with the requirements of Section II.2 of this plan. The applicant is permitted any of three alternates for deriving the fracture toughness of the disk materials.

3. High Temperature Properties

The high temperature properties of the high pressure rotor materials, including specimen test data, where applicable, are reviewed and compared with the requirements of Section II.3 of this plan.

4. Preservice Inspection

The preservice inspection program, including finish machining, ultrasonic inspection, surface inspection, visual inspection, and spin testing, is reviewed and compared with the requirements of Section II.4 of this plan. The extent to which the ultrasonic inspections and the acceptance criteria in the SAR agree with ASME Boiler and Pressure Vessel Code, Section III, NB-2530 for plate materials or NB-2540 for forgings, is reviewed.

5. Turbine Disk Design

The design and stress analysis procedures used for the high and low pressure turbine disks 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 at design overspeed and allowable stresses.

The SAR data are compared and evaluated against Section II.5 of this plan.

6. Inservice Inspection

The inservice inspection program described by the applicant, 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 failure of a turbine disk or rotor can be minimized by the use of suitable materials, adequate design, and preservice and inservice inspections. The applicant has described his program for assuring the integrity of low pressure turbine disks and high pressure turbine rotors by use of suitable materials of adequate fracture toughness, conservative design practices, and preservice and inservice inspections. The staff concurs that these provisions, as described in the SAR, provide reasonable assurance that the turbine disks will not fail during normal operation, including transients up to design overspeed."

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 III, V, and XI, American Society of Mechanical Engineers.

3. ASTM E-208, "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 Materials.

4. ASTM A-370, "Standard Methods and Definitions for Mechanical Testing of Steel Products," Annual Book of ASTM Standards, Parts 1, 2, 3, 4, or 31, American Society for Testing Materials.

5. J. A. Begley and W. A. Logsdon, Scientific Paper 71-1E7-MSLRF-P1, Westinghouse Electric Corp., July 26, 1971.

6. F. J. Witt and T. R. Mager, ORNL-TM-3894, Oak Ridge National Laboratory (1972).

SECTION 10.3 MAIN STEAM SUPPLY SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)
Power Systems Branch (PSB)

Secondary - Reactor Systems Branch (RSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Instrumentation and Control Systems Branch (ICSB)

I. AREAS OF REVIEW

The main steam supply system (MSSS) for both boiling water reactor (BWR) and pressurized water reactor (PWR) plants transports steam from the nuclear steam supply system to the power conversion system and various safety-related or nonsafety-related auxiliaries. Portions of the MSSS may be used as a part of the heat sink to remove heat from the reactor facility during certain operations and may also be used to supply steam to drive engineered safety feature pumps. The MSSS may also include provisions for secondary system pressure relief in PWR plants.

The MSSS for the BWR direct cycle plant extends from the outermost containment isolation valves up to and including the turbine stop valves, and includes connected piping of 2-1/2 inches nominal diameter and larger up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of reactor operation. The MSSS for the PWR indirect cycle plant extends from the connections to the secondary sides of the steam generators up to and including the turbine stop valves, and includes the containment isolation valves, safety and relief valves, connected piping of 2-1/2 inches nominal diameter and larger up to and including the first valve that is either normally closed or capable of automatic closure during all modes of operation and the steam line to the auxiliary feedwater pump turbine. The ASB is responsible for the review of the MSSS from the containment up to and including the outermost isolation valve. The PSB is responsible for the review of the remainder of the MSSS. (The turbine stop valve review is included in SRP Section 10.2). The PSB also determines the adequacy of the design, installation, inspection, and testing of the electrical power supplies for essential components required for proper operation of the MSSS. The design of the MSSS must be in accordance with General Design Criteria 2, 4 and 34.

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1. The ASB and PSB review the MSSS to determine which, if any, portions of the system are essential for safe shutdown of the reactor or for preventing or mitigating the consequences of accidents. The system is reviewed to verify that:

   a. A single malfunction or failure of an active component would not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.

   b. Appropriate quality group requirements and seismic design requirements are met for safety-related portions of the system.

   c. Failures of non-seismic Category I equipment or structures, or pipe cracks or breaks in high and moderate energy piping will not preclude essential functions of safety-related portions of the system.

   d. The system is capable of performing multiple functions such as transporting steam to the power conversion system, providing heat sink capacity or pressure relief capability, or supplying steam to drive safety system pumps (e.g., turbine-driven auxiliary feedwater pumps), as may be specified for a particular design.

   e. The design of the MSSS includes the capability to operate the atmospheric dump valves remotely from the control room following a safe shutdown earthquake coincident with the loss of offsite power so that a cold shutdown can be achieved with dependence upon safety grade components only.

2. The ASB reviews the MSSS with regard to measures provided to limit blowdown of the system in the event of a steam line break.

3. The ASB and PSB also review the design of the MSSS with respect to the following:

   a. The functional capability of the system to transport steam from the nuclear steam supply system as required during all operating conditions.

   b. The capability to detect and control system leakage, and to isolate portions of the system in case of excessive leakage or component malfunctions.

   c. The capability to preclude accidental releases to the environment.

   d. Provisions for functional testing for safety-related portions of the system.

4. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this SRP section.

Secondary reviews are performed by other branches and the results used by the ASB to complete the overall evaluation of the system. The secondary reviews are as follows. The RSB identifies essential components associated with the portion of the MSSS inside the primary containment that are required for normal operations and accident conditions, establishes shutdown cooling load requirements versus time, and verifies the design transient used in establishing the flow capacity and set point(s) of steam generator relief and safety valves. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB verifies 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 ICSB reviews portions of the MSSS with respect to the adequacy of design, installation, inspection, and testing of essential components necessary for instrumentation and control functions.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the MSSS, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides.

The design of the MSSS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion 2, as related to safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, with respect to safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 34, as related to the system function of transferring residual and sensible heat from the reactor system in indirect cycle plants.

4. Regulatory Guide 1.26, as related to the quality group classification of the system.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Regulatory Guide 1.102, as related to the protection of structures, systems, and components important to safety from the effects of flooding.

7. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles.

8. Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

9. Branch Technical Position RSB 5-1, as related to design requirements for residual heat 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 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 and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II. For review of 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.

The procedures for OL applications include 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 as a result of the staff's review.

Upon request from the primary reviewers, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewers obtain and use such input as required to assure that this review procedure is complete.

The review procedures below are written for typical MSSSs for both direct and indirect cycle plants. The reviewer will select and emphasize material from this SRP section, as may be appropriate for a particular case.

1. There are significant differences in the design of the MSSS for an indirect cycle (PWR) plant as compared to that for a direct cycle (BWR) plant. Further, different portions of the MSSS are safety-related in different plant designs, although the safety functions of the system are much the same in all PWR plants, and also in all BWR plants. The first step in the review of the MSSS, then, is to determine which portions are designed to perform a safety function. For this purpose, the system is evaluated to determine the components and subsystems necessary for achieving safe reactor shutdown in all conditions or for performing accident prevention or mitigation functions.

2. The reviewer determines that essential (safety-related) portions of the MSSS are correctly identified and are isolable to the extent required from non-essential portions of the system. The system description and piping and instrumentation diagrams (P&IDs) are reviewed to verify that they clearly indicate the physical division between each portion. System arrangement drawings are reviewed to identify the means provided for accomplishing system isolation.

3. The SAR is reviewed to verify that essential portions of the MSSS are designed to Quality Group B or seismic Category I requirements, and to verify that the design classifications specified meet the acceptance criteria. In general, the main steam lines from the steam generators to the containment isolation valves in PWR plants are classified seismic Category I and Quality Group B. The main steam lines in BWR plants from the outer containment isolation valves to the MSSS shutoff valves or the turbine stop valves are classified seismic Category I and Quality Group B.

4. The SAR is reviewed to assure that design provisions have been made to permit appropriate functional testing of system components important to safety. It is acceptable if the SAR delineates a testing and inspection program and the system drawings show any test recirculation loops and special connections around isolation valves that would be required by this program.

5. The system description, safety evaluation, component table, and P&IDs are reviewed to verify that the system has been designed to:

   a. Provide the necessary quantity of steam to any turbine-driven safety system pumps. The reviewer refers to the pump performance curves and verifies that the design is capable of providing the required steam flow to the turbine so that an adequate supply of water can be pumped. (OL)

   b. Assure safe plant operation by including appropriate design margins for pressure relief capacity and set points for the secondary system, and for removal of decay heat during various accident conditions, as may be applicable in a particular case. The review is done on a case-by-case basis, and system acceptability is based on a comparison of system flow rates, heat loads, maximum temperatures, and heat removal capabilities to those of similarly designed systems for previously reviewed plants. For PWRs the design is reviewed to verify system capability for controlled cooldown to about 350°F to allow actuation of RHR system.

   c. Provide leakage detection means for steam leakage from the system in the event of a steam line break. Temperature or pressure sensors are acceptable means for initiating signals to close the main steam line isolation valves and/or turbine stop valves to limit the release of steam during a steam line break accident.

d. Assure that in the event of a postulated break in a main steam line in a PWR plant, the design will preclude the blowdown of more than one steam generator, assuming a concurrent single component failure. In this regard the turbine stop and control valves are considered to be functional. The reviewer should verify that the main steam isolation valves and turbine stop and bypass valves can close against maximum steam flow.

e. Assure that in the event of a postulated safe shutdown earthquake in a PWR plant, the design includes the capability to operate atmospheric dump valves remotely from the control room so that cold shutdown can be achieved using only safety grade components, assuming a concurrent loss of offsite power (refer to Branch Technical Position RSB 5-1 attached to SRP Section 5.4.7).

6. The reviewer verifies that the system is designed so that essential functions will be maintained, as required, in the event of adverse environmental phenomena, certain pipe breaks, or loss of offsite power. The reviewer uses engineering judgment and the results of failure modes and effect analyses to determine that:

a. Failure of non-seismic Category I portions of the MSSS or of other systems located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the MSSS, do not preclude operation of the essential portions of the MSSS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that confirm that the above conditions are met are acceptable.

b. Essential portions of the MSSS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are evaluated under the SRP Section 3 series. The locations and the design of the system and structures are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of winds, flooding, and tornado missiles is acceptable.

c. Essential portions of the MSSS are protected from the effects of high and moderate energy line breaks and cracks, including pipe whip, jet forces and environmental effects. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the MSSS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in SRP Section 3.6.

d. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The SAR is reviewed to verify that for each MSSS component or subsystem affected by a loss of offsite power the system functional capability meets or exceeds minimum design requirements. Statements in the SAR and results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This is an acceptable verification of system functional reliability.

7. The descriptive information, P&IDs, MSSS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system will function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the analyses presented in the SAR to assure 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 accident situation for the required time spans. For each case the design is acceptable if minimum system requirements are met.

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 supply system (MSSS) includes all components and piping from the outermost containment isolation valves for BWRs and from the steam generator connection for PWRs and including the turbine stop valves. Based on the review of the applicant's proposed design criteria, design bases, and safety classification for safety-related portions of the MSSS and system performance requirements for normal, abnormal, and accident conditions, the staff concludes that the design of safety-related portions of the MSSS and auxiliary supporting systems is in conformance with the Commission's regulations as set forth in the General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," General Design Criterion 4, "Environmental and Missile Design Bases," General Design Criterion 34, "Residual Heat Removal," and meets the guidelines contained in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.29, "Seismic Design Classification," Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Regulatory Guide 1.117, "Tornado Design Classification," and Branch Technical Positions ASB 3-1 and MEB 3-1 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 34, "Residual Heat Removal."

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 1.102, "Flood Protection for Nuclear Power Plants."

7. Regulatory Guide 1.117, "Tornado Design Classification."

8. 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.

9. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," attached to SRP section 5.4.7.

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SECTION 10.3.6 STEAM AND FEEDWATER SYSTEM MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 1 requires that systems important to safety shall be designed to quality standards commensurate with the importance to safety of the functions to be performed.

The following areas relating to the general materials considerations for ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Class 1, 2 and 3 components of the steam and feedwater systems are reviewed: (The review procedures for materials considerations for steam generators are given in Standard Review Plan 5.4.2.1.) The Class 1 components of the steam and feedwater systems in boiling water reactors (BWRs) include those portions located between the reactor core and the outermost containment isolation valves. The Class 2 and 3 components of the steam and feedwater systems include those portions beyond the outermost containment isolation valves in boiling water reactors (BWRs) and the secondary coolant system of pressurized water reactors (PWRs), except for those portions of the steam generator that come in contact with the primary coolant.

1. Fracture Toughness of Class 1, 2 and 3 Components

The fracture toughness properties and requirements for Class 1, 2 and 3 components are reviewed. Typical components in this review include steam generator shells in PWRs, as well as carbon or low alloy steel portions of steam and feedwater lines in both PWRs and BWRs.

2. Materials Selection and Fabrication

a. The materials selected for all Class 1, 2 and 3 components and their fabrication are reviewed.

b. For austenitic stainless steel components, the following points are reviewed:

(1) The steps taken to control the use of sensitized stainless steel.

(2) The controls placed on the composition of any nonmetallic external thermal insulation.

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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.

(3) The controls placed on the welding procedures.

c. For all components the following points are reviewed:

(1) The qualification procedures for welds in areas of limited accessibility are reviewed.

(2) The cleaning procedures are reviewed.

(3) For tubular products, the nondestructive examination procedures are reviewed for conformance with the ASME Code.

d. For carbon or low alloy steel components, the controls placed on the welding procedures are reviewed.

II. ACCEPTANCE CRITERIA

1. Fracture Toughness of Class 1, 2 and 3 Components

The fracture toughness properties of the ferritic materials of these components must meet the following requirements of the Code, Section III:

NB-2300 - "Fracture Toughness Requirements for Materials" (Class 1)
NC-2300 - "Fracture Toughness Requirements for Materials" (Class 2)
ND-2300 - "Fracture Toughness Requirements for Materials" (Class 3)

2. Materials Selection and Fabrication

a. (1) The materials specified for use in Class 1, 2 and 3 components must conform to Appendix I to Section III of the Code, and to Parts A, B, and C of Section II of the Code.

(2) 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.

b. The following criteria are applicable to all austenitic stainless steel components:

(1) Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," describes acceptable criteria for preventing stress corrosion of stainless steel components of the steam and feedwater systems. Furnace-sensitized material should not be used, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling and storage, testing materials prior to fabrication, and determining the degree of sensitization that occurs during welding. Pickling should not be performed on sensitized stainless steel.

(2) The composition of nonmetallic thermal insulation for the austenitic stainless steel components should be controlled as described in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Concentrations of leachable contaminants should be controlled as specified in position C.2.b and Figure 1 of this guide, especially with regard to sodium silicate inhibitor concentrations, to reduce the probability of stress corrosion cracking of these components when the insulation is moistened. Nonmetallic thermal insulation applied to nonaustenitic components should also be controlled as above. When the insulation is wet, the drippings can affect austenitic stainless steel at a lower elevation.

(3) Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," describes acceptable criteria for assuring the integrity of welds in stainless steel components.

c. The following criteria are applicable to all components:

(1) Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for assuring the integrity of welds in locations of restricted direct physical and visual accessibility.

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

(b) Requalification is required for different accessibility conditions or when other essential variables listed in the Code, Section IX, are changed.

(c) The qualification and requalification tests required by 2.c.(1)(a) and (b) above may be waived provided that the joint is to be 100% radiographed or ultrasonicly examined after completion of the weldment. Examination procedures and acceptance standards should meet the requirements of the ASME Code Section III. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance documentation of the completed weld.

(2) Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," describe acceptable procedures for cleaning and handling Class 2 and 3 components of the steam and feedwater systems. 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 in these vented tanks need not be controlled.

(3) Acceptance criteria for nondestructive examination of tubular products are given in the ASME Code, Section III, Paragraphs NB/NC/ND 2550 through 2570.

d. The following criterion is applicable to all carbon or low alloy steel components:

(1) Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," describes acceptable criteria for supplementing Section IX of the Code, to prevent crack formation in the underbead areas and heat-affected zones of welds in these materials. The welding procedures should be qualified at the minimum preheat temperature, and production welds should be monitored to verify that the limits on preheat and interpass temperature are maintained.

(2) The preheat temperature controls described in the Westinghouse topical report WCAP-8577 are an acceptable alternate to compliance with position C.2 of Regulatory Guide 1.50.

Preheat recommendations in Section III, Article D-1000 of the Code shall apply unless an alternate procedure is justified.

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 areas listed in Section I for the required information, in accordance with the following procedures:

1. Fracture Toughness of Class 1, 2 and 3 Components

   The reviewer determines which components of the steam and feedwater systems will be made of carbon or low alloy steels, and determines that their fracture toughness properties are in conformance with Section II.1.

2. Materials Selection and Fabrication

   The reviewer determines that the materials proposed for the steam and feedwater systems are in conformance with Appendix I to Section III and to parts A, B, or C of Section II of the Code.

   a. For austenitic stainless steel components, the following procedures are followed:

      (1) The reviewer examines the methods of controlling sensitized stainless steel and determines that they comply with the acceptance criteria stated in Section II.2.b.(1), especially with respect to cleaning and protection from contamination during handling and storage, verification of nonsensitization of the material, and qualification of welding processes employed in production.

      If alternative methods of testing 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 and on MTEB positions taken on previous applications. The reviewer may ask the applicant to justify technically his departures from the above-cited criteria. Alternative tests of qualification welds that have been previously accepted by the MTEB include the use of ASTM A-708 for determining the sensitization of the heat affected zones of the qualification welds, for qualifying welds, and testing raw materials for nonsensitization.

      (2) The reviewer determines whether nonmetallic thermal insulation will be used on any austenitic stainless steel components of the steam and feedwater systems, and verifies that the leachable impurities in this insulation lie within the "acceptable analyses" area of Figure 1 of Regulatory Guide 1.36, as discussed in Section II.2.b.(2).

      (3) The reviewer examines the methods of controlling and measuring the amount of delta ferrite in stainless steel weld deposits, in accordance with the criteria stated in Section II.2.b.(3), especially with respect to the filler metal acceptance procedures for the determination of delta ferrite content.

   b. The reviewer determines that the methods for qualifying welders for making welds in locations of restricted direct physical and visual accessibility,

and the methods for monitoring and certification of production welds in such areas are in accordance with the acceptance criteria stated in Section II.2.c.(1).

c. The reviewer determines that the methods for cleaning and handling the Class 1, 2 and 3 components are in accordance with acceptance criteria stated in Section II.2.c.(2).

d. For all carbon or low alloy steel components, the reviewer verifies that the minimum preheat and interpass temperatures for welding are specified in accordance with Section II.2.d.

e. The reviewer verifies that the tubular products are examined in accordance with acceptance criteria stated in Section II.2.c.(3).

3. General

If the information contained in the safety analysis report 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 OF 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 mechanical properties of materials selected for Class 1, 2 and 3 components of the steam and feedwater systems satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, and Parts A, B, or C of Section II of the Code."

"The fracture toughness properties of ferritic materials satisfy the requirements of the Code." "These fracture toughness tests and mechanical properties required by the Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture."

"The controls imposed upon austenitic stainless steel are in accordance with 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." Fabrication and heat treatment practices performed in accordance with these recommendations provide reasonable assurance that microfissuring and stress corrosion cracking will not occur during the design life of the plant. The controls placed upon concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the steam and feedwater systems are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

"The welding procedures used in limited access areas conform to Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The onsite cleaning and cleanliness controls during fabrication satisfy the positions given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants." The precautions taken in controlling and monitoring the preheat and interpass temperatures during welding of carbon and low alloy steel components conform to Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel."

"Tubular Products are nondestructively examined in accordance with the Code."

"Conformance with the codes, standards, and Regulatory Guides mentioned constitutes an acceptable basis for assuring the integrity of steam and feedwater systems, and for meeting in part the requirements of General Design Criterion 1."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code: Section III, Subsection NB, NC, and ND, Appendix I and Appendix G; Section II, Parts A, B, and C; and Section IX; American Society of Mechanical Engineers.

3. ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

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

5. ASTM A-708, Detection of Susceptibility of Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel," Annual Book of ASTM Standards, American Society for Testing and Materials.

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 Nuclear Power Plants."

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

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

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

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

13. Standard Review Plan 5.4.2.1, "Steam Generator Materials."

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SECTION 10.4.1 MAIN CONDENSERS

REVIEW RESPONSIBILITIES

Primary - Power Systems Branch (PSB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system.

1. The PSB reviews the performance requirements of the main condenser for both direct and indirect cycle plants during all operating conditions. Emphasis will be placed on the review of direct cycle facilities with regard to the prevention of loss of vacuum, corrosion and/or erosion, and hydrogen buildup.

2. The PSB reviews the design of the MC system with respect to the following:

   a. The means to detect, control and facilitate correction of the leakage of cooling water into the condensate; to detect radioactive leakage into or out of the system; and to preclude accidental releases of radioactive materials to the environment in amounts in excess of established limits.

   b. Instrumentation and control features that determine and verify that the MC is operating in a correct mode.

   c. The means provided to deal with flooding from a complete failure of the MC and to preclude damage to safety-related equipment from the flooding.

   d. The capability of the MC to withstand the blowdown effects of steam from the turbine bypass system.

Secondary reviews are performed by other branches and the results used by the PSB to complete the overall evaluation of the system. The secondary reviews are as follows. The ETSB evaluates the inventory of radioactive contaminants in the MC during power operation and during shutdown. The MTEB evaluates the adequacy of the materials of construction, the methods used to reduce the corrosion and/or erosion of MC tubes and

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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.

components, the permissible cooling water inleakage, and the allowed time of operation with inleakage without affecting condensate/feedwater quality for safe reactor operation.

II. ACCEPTANCE CRITERIA

There are no General Design Criteria that directly apply to the main condenser. Acceptability of the design of the main condenser system, as described in the applicant's safety analysis report (SAR), is based on the system being designed such that failures do not cause unacceptable condensate quality, flooding of areas housing safety-related equipment, or result in excessive releases of radioactivity to the environment. Additional bases for determining the acceptability of the MC system are the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience, and Regulatory Guide 1.68, as related to preoperational and startup testing of the power conversion system.

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 and bases and the preliminary design meet the acceptance criteria given in subsection II. For the review of 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. 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 SAR is reviewed to determine that the system description delineates the main condenser system capabilities including the minimum system heat transfer and system flow requirements for normal plant and turbine bypass operation. Measures provided to prevent loss of vacuum, corrosion and/or erosion of MC tubes and components, and hydrogen buildup in the MC are reviewed, with particular emphasis on these measures for direct cycle (boiling water reactor) plants. System performance requirements are reviewed to determine that they satisfactorily limit possible system degradation conditions (e.g., leakage, partial loss of vacuum) and describe the procedures that are followed to detect and correct these conditions. The SAR is also reviewed to determine that any allowed MC system degraded operation does not have an adverse effect on the reactor primary system or secondary system in the case of pressurized water reactors.

2. The reviewer evaluates the MC system design to verify that:

   a. Means have been provided for detecting, controlling and correcting condenser cooling water leakage into the condensate.

   b. The permissible cooling water inleakage and time of operation with inleakage are provided to assure that condensate/feedwater quality can be maintained within safe limits.

   c. Measures have been provided to detect radioactive leakage into and out of the MC system and to preclude unacceptable accidental releases of radioactivity to the environment from the system.

   d. The system is provided with instrumentation and control features that determine and verify that the MC is operating in a correct mode.

3. The reviewer uses engineering judgment and the results of failure modes and effects analyses to determine that:

   a. The failure of a main condenser and the resulting flooding will not preclude operation of any essential systems. Reference to sections of the SAR describing plant features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable.

   b. The system, in conjunction with the main steam system, has provisions to detect loss of condenser vacuum and to effect isolation of the steam source. For direct cycle plants, it will be acceptable if the detection system in the MC can actuate the main steam isolation valves to limit the quantity of steam lost from the condenser.

   c. Design provisions have been incorporated into the MC that will preclude component or tube failures due to steam blowdown from the turbine bypass system.

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 condenser system includes all components and equipment from the turbine exhaust to the connections and interfaces with the main condensate and other systems. The scope of review of the main condenser system for the _____ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the main condenser system and supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the main condenser system and the requirements to

preclude safety-related equipment malfunctions or failures due to rupture of the main condensers. (CP)] [The review has determined that the design of the main condenser system and supporting systems is in conformance with the proposed design criteria and design bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's design criteria and design bases for the main condenser system and supporting systems to applicable regulations, staff technical positions and industry standards.

"The staff concludes that the design of the main condenser system conforms to all applicable regulations, staff positions and industry standards, and is acceptable."

V. REFERENCES

1. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

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SECTION 10.4.2 MAIN CONDENSER EVACUATION SYSTEM

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.

1. The main condenser evacuation system (MCES) generally consists of two subsystems: the "hogging" or startup system which initially establishes main condenser vacuum, and the normal system which maintains condenser vacuum once it has been established. The review of each MCES subsystem includes the design, design objectives, capacity, method of operation, and factors that influence gaseous radioactive material handling, e.g., system interfaces and potential bypass routes. The ETSB review includes the system piping and instrumentation diagrams (P&IDs).

2. The quality group classifications of piping and equipment, and the bases governing the design criteria chosen are reviewed.

3. Design features to preclude the possibility of an explosion if the potential for explosive mixtures exists are reviewed.

Provisions incorporated to sample and monitor radioactive materials in gaseous effluents from the MCES are reviewed in SRP Sections 11.5 and 9.3.2.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. The MCES capacity should be consistent with the industry guidelines given in Reference 2. Either mechanical vacuum pumps or steam jet air ejectors may be used for hogging (startup) or normal evacuation of the main condenser.

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2. The components of the system may be designed to Quality Group D as defined in Regulatory Guide 1.26 (Ref. 3) and to a nonseismic design classification.

3. If there is a potential that explosive mixtures may exist, the MCES should be designed to withstand the effects of an explosion or provide redundant instrumentation to detect and annunciate the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, II Acceptance Criteria, Item 6.

4. Provisions to control and monitor releases of radioactivity to the environment from the MCES must conform to the requirements of General Design Criteria 60 and 64 (Ref. 1).

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this SRP section as may be appropriate for a particular case.

1. In the ETSB review of the MCES, the P&IDs are reviewed to determine the flow paths of gases through the system, including all bypasses, and the points of release of gaseous wastes to the environment or other systems. This information is used in SRP Section 11.3 to calculate the quantity of radioactive material released annually in gaseous effluents during normal operations, including anticipated operational occurrences. ETSB verifies that water from the mechanical vacuum pumps and condensate from the steam jet air ejectors are classified as radioactive liquids and treated accordingly.

2. ETSB reviews the equipment quality group classifications.

3. If there is a potential that explosive mixtures may exist, ETSB determines whether the applicant has designed the MCES to withstand the effects of such an explosion, or has provided instrumentation to detect, annunciate, and prevent the buildup of potentially explosive mixtures. ETSB will also determine if the applicant's design includes adequate provisions to stop continuous leakage paths after an explosion.

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 main condenser evacuation system includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment. The scope of our review included the system capability to transfer radioactive gases to the gaseous waste processing system or ventilation exhaust systems, the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in

accordance with General Design Criteria 60 and 64 and the quality group classification of equipment and components used to collect gaseous radioactive wastes relative to the guidelines of Regulatory Guide 1.26. We have reviewed the applicant's system descriptions, piping and instrumentation diagrams, and design criteria for the components of the main condenser evacuation system. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the main condenser evacuation system to the applicable regulations, regulatory guides and industry standards referenced above. Based on our evaluation, we find the proposed main condenser evacuation system acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and Criterion 64, "Monitoring Radioactivity Releases."

2. "Standards for Steam Surface Condensers," 6th Ed., Heat Exchanger Institute (1970).

3. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 10.4.3 TURBINE GLAND SEALING SYSTEM

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.

The turbine gland sealing system design, design objectives, method of operation, and factors that influence gaseous radioactive material handling, e.g., source of sealing steam. system interfaces, and potential leakage paths are reviewed. The ETSB review includes piping and instrumentation diagrams (P&IDs).

Provisions to sample and monitor radioactive materials in gaseous effluents are reviewed in SRP Sections 11.5 and 9.3.2.

The quality group classification of piping and equipment and the bases of the design criteria selected are reviewed.

Provisions for controlling the release of radioactive materials from the gland seal condenser vent are reviewed in SRP Section 11.3.

During the OL stage, the ASB reviews the potential effect of high energy pipe breaks within this system on safety-related equipment.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

The turbine gland sealing system should be designed to provide for the collection and condensation of sealing steam and the venting and treatment (as required in Ref. 1) of noncondensables. Quality Group D as defined in Regulatory Guide 1.26 (Ref. 2) and a nonseismic design classification are acceptable design criteria for this system.

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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.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from this SRP section, as may be appropriate for a particular case.

ETSB reviews the equipment quality group classification.

ETSB reviews the system P&IDs to determine the source of sealing steam and the disposition of steam and noncondensables vented from the gland seal. Where sealing steam from primary coolant condensate is used, the review includes the radiological processing and monitoring provisions in accordance with SRPs 11.3, 11.5 and 9.3.2.

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 turbine gland sealing system includes the equipment and instruments to provide a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The scope of our review included the source of sealing steam, and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with General Design Criteria 60 and 64. We have reviewed the applicant's system descriptions and design criteria for the components of the turbine gland sealing system and found them consistent with Regulatory Guide 1.26.
>
> "The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the turbine gland sealing system to the applicable regulations and regulatory guides referenced above. Based on our evaluation, we find the proposed turbine gland sealing system acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Material to the Environment," and Criterion 64, "Monitoring Radioactivity Releases."

2. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

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SECTION 10.4.4 TURBINE BYPASS SYSTEM

REVIEW RESPONSIBILITIES

Primary - Power Systems Branch (PSB)

Secondary - Reactor Systems Branch (RSB)
Auxiliary Systems Branch (ASB)

I. AREAS OF REVIEW

The turbine bypass system (TBS) provides operational flexibility so that the plant may accept certain load changes without disturbing the nuclear steam supply system. The TBS is designed to discharge a stated percentage of rated main steam flow directly to the main condensers, bypassing the turbine. This steam bypass enables the plant to take step load reduction up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure for a boiling water reactor (BWR) and steam generator pressure for a pressurized water reactor (PWR). The TBS is not required for safe shutdown as the relief and safety valves are operated under emergency conditions. The system is not required to function as a heat sink for the prevention or mitigation of postulated accidents. Failure of the TBS during a load reduction or turbine trip would result in the actuation of the relief valves and possibly the safety valves.

The PSB reviews the system from the branch connection at the main steam system to the main condensers.

1. The PSB reviews the TBS to determine that a failure of the system or system components will not have an adverse effect on essential equipment.

2. The PSB reviews the TBS functional requirements for both normal and abnormal operating conditions, and with respect to the following: (a) capability to isolate those portions of the system that could leak or malfunction; (b) capability to perform adequate operational testing and inservice inspection; (c) to assure there are no adverse effects of postulated system piping failures on safety-related equipment; and (d) to reduce the possibility of reactor transients due to inadvertent operation of the TBS from faults in the TBS instrumentation and control.

3. The applicant's proposed technical specifications are reviewed at the operating license stage, as they relate to areas covered in this SRP section.

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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.

Secondary reviews are performed by other branches and the results are used by the PSB to complete the overall evaluation of the system. The secondary reviews are as follows. The ASB determines that the appropriate seismic and quality group classifications have been established for system components and also that TBS is in accordance with Branch Technical Positions ASB 3-1 and MEB 3-1 as related to cracks and breaks in high and moderate energy piping outside of containment. The RSB determines that the steam bypass capacity is consistent with reactor transient analysis.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the turbine bypass system, as described in the applicant's safety analysis report (SAR), is based on the criteria below. An additional basis for determining the acceptability of the TBS will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of the TBS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. Failure or malfunction of the TBS does not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).

2. Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

3. Regulatory Guide 1.68, as related to preoperational and startup testing of the power conversion system.

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 and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II. For review of 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.

The procedures for OL applications include 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 as a result of the staff's review.

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.

1. The SAR is reviewed to determine that the system description and piping and instrumentation diagrams (P&IDs) delineate the system and components.

2. The SAR is reviewed to verify that the system design bases and an evaluation of the system capacity are provided, including the relation between the TBS capacity and relief valve capacity in terms of percentage of rated main steam flow, the maximum reactor power step change the system is designed to accommodate without a reactor or turbine trip, and the maximum electric load step change the reactor is designed to accommodate without reactor control rod motion or steam bypassing.

3. The reviewer uses engineering judgment and the results of failure modes and effects analyses to determine that:

   a. Failure of the TBS to operate will not preclude operation of any essential systems. Statements in the SAR that confirm the above are acceptable.

   b. Failure of the TBS high energy piping will not have adverse effects on any safety-related systems or components that may be located close to the system.

IV. EVALUATION FINDINGS

The reviewer determines 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 turbine bypass system (TBS) includes all components and piping from the branch connection at the main steam system to the main condensers. The scope of review of the turbine bypass system for the _____ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the TBS and auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the turbine bypass system and the requirements for safe operation of the TBS during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the turbine bypass system and auxiliary supporting systems is in conformance with the design criteria and 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 turbine bypass system and its supporting systems to applicable regulations, staff technical positions and industry standards.

"The staff concludes that the design of the turbine bypass system conforms to all applicable regulations, staff positions and industry standards and is acceptable."

V. REFERENCES

1. 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.

2. Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants."

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SECTION 10.4.5 CIRCULATING WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Instrumentation and Control Systems Branch (ICSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems.

1. The ASB reviews the performance of the CWS with respect to its functional requirements and the effects of adverse environmental occurrences, abnormal operational transients, or accident conditions such as loss of offsite power.

2. The ASB reviews the CWS and its interfaces with other systems to determine that a malfunction, failure of a component, or failure of a circulating water pipe including the failure of an expansion joint do not have unacceptable adverse effects on the functional performance capabilities of safety-related systems located in the immediate area.

3. ASB further reviews the design of the circulating water system with respect to the following:

   a. The capability to prevent or detect and control flooding of safety-related areas so that the intended safety function of a safety system or component will not be precluded due to circulating water system leakage.

   b. The compatibility of the methods proposed for control of water chemistry and of long-term corrosion and organic fouling with system components and piping materials.

   c. Provisions for instrumentation to permit operational testing of the system and to annunciate abnormal and unsafe operating conditions.

Secondary reviews are performed by other branches and the results used by the ASB to complete the overall evaluation of the system. The secondary reviews are as follows:

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The ICSB and PSB will, upon request, review the system electrical components, instrumentation and controls as they may relate to operations that could effect safety-related systems or components.

II. ACCEPTANCE CRITERIA

Acceptability of the circulating water system, as described in the applicant's safety analysis report (SAR), is based on the criteria below. There are no general design criteria or regulatory guides that directly apply to the functional performance requirements for the CWS. Specific criteria for the CWS are as follows:

1. Means should be provided to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS.

2. Malfunction or a failure of a component or piping of the CWS including an expansion joint should not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.

3. Agents used for the control of water chemistry, corrosion, and organic fouling should be compatible with the materials of the system.

4. Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

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 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 review of 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.

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 will select and emphasize material from this SRP section as may be appropriate for a particular case.

1. Although the circulating water system is not safety-related, a failure of this system, or any of its components, may affect a safety-related component or system.

Since large quantities of water flow through the CWS, a leak or break in a component or pipe or expansion joint failure could cause severe and unacceptable flooding of adjacent areas. In evaluating the effects of the failure of an expansion joint, ASB assumes that the butterfly valve(s) are not available to isolate CWS flow out of the failed expansion joint unless the valve(s) have been designed to safety grade requirements. The ASB reviews the descriptions and drawings in the SAR and determines that provisions are incorporated in the design to prevent unacceptable flooding of areas containing safety-related equipment or to mitigate the consequences of flooding.

2. The ASB reviews the CWS to verify that the capability to detect leaks and secure the system quickly and effectively exists. The reviewer verifies that the design includes provisions to minimize hydraulic transients and their effect upon the functional capability and the integrity of system components.

3. Based on the information contained in the SAR, the reviewer verifies that the applicant's proposed methods for control of water chemistry and of long-term corrosion and organic fouling, and the chemical agents used for these purposes are compatible with the system materials.

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 circulating water system includes all components and equipment necessary to provide the main condenser with a continuous supply of cooling water. Based on the review of the applicant's proposed design criteria and bases for the circulating water system, the staff concludes that the design of the circulating water system is in conformance with the design criteria and design bases.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the circulating water system to applicable staff positions and industry standards.

"The staff concludes that the design of the circulating water system conforms to all applicable staff positions and industry standards, and is acceptable."

V. REFERENCES

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.

SECTION 10.4.6 CONDENSATE CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The condensate cleanup system (CCS) removes dissolved and suspended impurities due to corrosion from condenser or steam generator leaks that could be introduced into the CCS by carryover from the main steam system. The CCS is not required for safe shutdown or mitigation of postulated accidents, but is important in maintaining the primary coolant quality in direct cycle plants or the secondary coolant quality in indirect cycle plants.

The ASB reviews the CCS from the supply point downstream of the condensate pumps to the discharge point upstream of the feedwater heaters, and also to the interfaces with the effluent treatment systems.

1. The ASB assures that the CCS is reviewed to verify that the system provides feedwater to the reactor for direct cycle plants or to steam generators for indirect cycle plants that meets water purity requirements. Condensers are reviewed to determine that proper design measures are taken to assure that the chloride concentration and other contaminant concentrations are limited to allowable values until the condensate and feedwater systems can be isolated in the event of condenser tube leaks.

2. ASB reviews the system to determine that the design satisfies the recommendations of Branch Technical Position ASB 3-1 with respect to breaks and cracks in high and moderate energy system piping.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this SRP section.

4. ETSB determines the effect of the CCS on fission and corrosion product concentrations and the effect of the quantity of spent resin and regenerant solution on radwaste system requirements as described in SRP Section 11.2.

Secondary review is performed by MTEB to determine that the quality of the water supplied to the reactor for direct cycle plants, and to the steam generators for indirect cycle

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plants, meets water purity requirements. MTEB reviews are performed to verify the compatibility of the materials of construction with the service conditions. MTEB will perform reviews to verify that the inservice inspection program requirements are met for the system components.

II. ACCEPTANCE CRITERIA

Acceptability of the CCS design, as described in the applicant's safety analysis report (SAR), is based on the criteria below. There are no general design criteria that apply directly to the CCS.

1. For direct cycle plants, the design of the CCS should conform to the recommendations of Regulatory Guide 1.56, as related to the design of condensate demineralizer systems to maintain the proper water purity specified for the reactor. For indirect cycle plants the design should conform with Positions C.1 through C.5 of Regulatory Guide 1.56 and the secondary water chemistry requirements as specified in the plant Technical Specifications.

2. The design of the CCS should meet Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

3. Inservice inspection requirements should conform to those in ASME Boiler and Pressure Vessel Code, Section XI for Class 2 and 3 components, as appropriate.

For those areas of review identified in subsection I of this SRP section as being the responsibility of MTEB, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to the MTEB.

III. REVIEW PROCEDURES

The review procedures 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 preliminary safety analysis report (PSAR) meet the acceptance criteria given in subsection II. For the review of 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 (FSAR).

Upon request from the primary reviewer, the MTEB will provide input for the areas of review stated in subsection I. The primary reviewer uses such input as required to assure that this review procedure is complete.

The procedures for OL reviews include 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 as a result of the staff's review. The reviewer will select and emphasize material from the paragraphs below, as may be appropriate for a particular case.

1. Layout drawings are reviewed to ascertain there are no safety-related components or systems located close to the high or moderate energy pipes of the CCS, or that protection from the effects of failures of these pipes is provided. The means of providing such protection will be given in Chapter 3.6 of the SAR and procedures for reviewing this information are given in SRP Section 3.6.

2. The ASB evaluates the system design information and drawings and, utilizing engineering judgment, operational experience, and performance characteristics of similar, previously approved systems, verifies that:

   a. The system meets the requirements for condensate cleanup capacity, provides effluent of the required purity, and contains adequate instrumentation to monitor the effectiveness of the system.

   b. The system has adequate capacity to maintain acceptable chloride concentrations and other contaminant concentrations in the feedwater system for a sufficient period of time to allow remedial action in the event of condenser tube failures, and the technical specifications and operational procedures adequately describe the method used to isolate the condensate and feedwater systems during repairs.

   c. The system is connected to radioactive waste disposal systems to allow disposal of spent resin or regenerant solutions when required.

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 condensate cleanup system includes all components and equipment necessary for the removal of dissolved and suspended impurities which may be present in the condensate. Based on the review of the applicant's proposed design criteria and design bases for the condensate cleanup system and the requirements for operation of the system, the staff concludes that the design of the condensate cleanup system and supporting systems is in conformance with the design criteria and design bases. (OL)

"The basis for acceptance in the staff review has been conformance of the applicant's design criteria and design basis for the condensate cleanup system and supporting systems to applicable regulatory guides, staff technical positions, and industry standards.

"The staff concludes that the design of the condensate cleanup system conforms to all applicable guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

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

2. 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.

3. ASME Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Plant Components."

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SECTION 10.4.7 CONDENSATE AND FEEDWATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Instrumentation and Control Systems Branch (ICSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The condensate and feedwater system (CFS) provides feedwater at the required temperature, pressure, and flow rate to the reactor for boiling water reactor (BWR) plants and to the steam generators for pressurized water reactor (PWR) plants. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low pressure feedwater heaters to the feedwater pumps, and then is pumped through the high pressure feedwater heaters to the nuclear steam supply system.

ASB reviews the CFS from the condenser outlet to the connection with the nuclear steam supply system and to the heater drain system to assure conformance to General Design Criteria 2, 4, 5, 44, 45 and 46. For indirect cycle plants, there are also interfaces with the secondary water makeup system and the auxiliary feedwater system. The CFS is used for normal shutdown. The only part of the CFS classified as safety-related, i.e., required for safe shutdown or in the event of postulated accidents, is the feedwater piping from the steam generators for PWRs and from the nuclear steam supply system for BWRs, up to and including the outermost containment isolation valve.

1. The ASB reviews the characteristics of the CFS with respect to the capability to supply adequate feedwater to the nuclear steam supply system as required for normal operation and shutdown.

2. The ASB review determines that an acceptable design has been established for:

   a. The interfaces of the CFS with the auxiliary feedwater system (PWR), the reactor core isolation cooling system (BWR), and the condensate cleanup system.

   b. The feedwater system (PWR), including the auxiliary feedwater system piping entering the steam generator, with regard to possible fluid flow instabilities

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(e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

c. The detection of major system leaks that could affect the functional performance of safety-related equipment.

d. System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from the effects of internally- and externally-generated missiles and from the effects of adverse environmental occurrences.

e. The system satisfies the recommendations of Branch Technical Position ASB 3-1 with respect to the effects of pipe whip and jet impingement that may result from high or moderate energy piping breaks or cracks (in this regard the CFS is considered to be a high energy system).

3. The ASB reviews the applicant's proposed technical specifications for operating license applications as they relate to areas covered in this SRP section.

4. The RSB determines that transients resulting from feedwater flow control malfunctions will not violate the primary system pressure boundary integrity criterion in accordance with SRP Sections 15.1.1 through 15.1.4, and that the loss of normal feedwater flow will not violate the fuel damage criterion or the system pressure boundary integrity criterion in accordance with SRP Section 15.2.7.

Secondary review evaluations are performed by other branches and the results used by the ASB to complete the overall evaluation of the system. The secondary reviews are as follows. The ICSB will, upon request, review the instrumentation and controls associated with the feedwater control system (BWR) or steam generator level control system (PWR). Upon request, the SEB determines the acceptability of design analyses, procedures, and criteria used to establish the structural adequacy of devices or restraints as they may relate to significant water hammer forces in system piping, the MEB reviews test programs and the operability of components that may by affected by water hammer and confirms that piping and components are designed in accordance with applicable codes and standards, and the MTEB verifies that inservice inspection requirements are met for system components that may be affected by forces from water hammer. The PSB evaluates the system power sources with respect to their capability, capacity, and reliability to perform safety-related functions during normal, transient, and accident conditions.

II. ACCEPTANCE CRITERIA

Acceptability of the condensate and feedwater system, as described in the applicant's safety analysis report (SAR), is based on specific General Design Criteria and regulatory guides. Listed below are the specific criteria as they relate to the CFS.

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, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. General Design Criterion 44, to assure:

   a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

   b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)

   c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

5. General Design Criterion 45, as related to design provisions to permit periodic inservice inspection of system components and equipment.

6. General Design Criterion 46, as related to design provisions to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

7. Regulatory Guide 1.26, as related to the quality group classification of safety-related system components.

8. Regulatory Guide 1.29, as related to the seismic design classification of safety-related system components.

9. Regulatory Guide 1.68.1, as related to the preoperational and initial startup testing of feedwater and condensate systems for boiling water reactor power plants.

10. Regulatory Guide 1.102, as related to the protection of structures, systems, and components important to safety from the effects of flooding.

11. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles.

12. Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

13. Branch Technical Position ASB 10-2, as related to the design guidelines to reduce the potential for water hammer in steam generators with top feedring designs (attached).

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 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 review of 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.

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 inputs as required to assure that this review procedure is complete.

The procedures for OL applications include 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 as a result of the staff's review.

The reviewer will select and emphasize material from this SRP section as may be appropriate for a particular case.

The SAR is reviewed to determine that the system description and diagrams delineate the function of the condensate and feedwater system under normal and abnormal conditions. The reviewer verifies the following:

1. The system has been designed to function as required for all modes of operation. The results of failure modes and effects analyses presented in the SAR, if any, are used in making this determination.

2. The system piping is designed to preclude hydraulic instabilities from occurring in the piping for all modes of operation. As appropriate, the reviewer evaluates the results of model tests and analyses that are relied on to verify

that water hammer will not occur, or proposed tests of the installed system that are intended to verify design adequacy. Steam generators that use top feed designs are reviewed in accordance with Branch Technical Position 10-2.

3. The outermost containment isolation valves and all downstream piping to the nuclear steam supply system are designed in accordance with seismic Category I and appropriate quality group requirements.

4. Breaks in system components or piping will not result in adverse effects on the functional performance of essential systems or components. The means for providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in SRP Section 3.6.

5. The CFS design is such that the plant can be safely shut down using the auxiliary feedwater system or the reactor core isolation cooling system, if required.

6. The CFS design, or other plant systems, provide the capability to detect and control leakage from the system.

7. Measures will be taken, as appropriate, to protect personnel from any toxic effects of chemicals used for feedwater treatment.

8. The essential portions of the CFS are protected from the effects of floods, hurricanes, tornadoes, and internally- or externally-generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the SRP Section 3 series. The location and design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado, missile and flood protected, or the components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

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 condensate and feedwater system includes all components and equipment from the condenser outlet to the connection with the nuclear steam supply system and to the heater drain system [secondary water makeup system, and auxiliary feedwater system interfaces. (PWRs only)]. Based on the review of the applicant's proposed design criteria, the design bases, and safety classification for the safety-related portions of the condensate and feedwater system and the requirements for system performance for all conditions of plant operation, the staff concludes that the design of the

condensate and feedwater system and supporting systems is in conformance with the Commission 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 5, "Sharing of Structures, Systems, and Components," General Design Criterion 44, "Cooling Water," General Design Criterion 45, "Inspection of Cooling Water System," and General Design Criterion 46, "Testing of Cooling Water System," and meets the guidelines contained in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.29, "Seismic Design Classification," Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems For Boiling Water Reactor Power Plants," Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Regulatory Guide 1.117, "Tornado Design Classification," Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammer in Steam Generators with Top Feedring Designs," and Branch Technical Positions ASB 3-1 and MEB 3-1, 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 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

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. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants."

10. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

11. Regulatory Guide 1.117, "Tornado Design Classification."

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.

13. Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammer in Steam Generators with Top Feedring Designs."

BRANCH TECHNICAL POSITION ASB 10-2
DESIGN GUIDELINES FOR WATER HAMMERS IN STEAM GENERATORS WITH TOP FEEDRING DESIGNS

BACKGROUND

Experience over the past decade or so has shown that top-feed steam generators containing feedwater spargers with bottom drain holes incur steam condensation induced water hammers. This type of water hammer has frequently occurred after the feedwater sparger was uncovered (due to some plant transient) and cold auxiliary feedwater flow was subsequently initiated. The initiation of the auxiliary feedwater flow into the steam generator produces a water slug in the sparger or feedwater piping, which is then accelerated by the unbalanced pressures produced by the condensation of a steam pocket in the line. The resultant impulse could be of a sufficient magnitude to cause damage to the steam generator or associated piping. An increasing number of these steam generator water hammer events has been occurring, in some cases resulting in negligible effects, and in others resulting in damage to piping support systems.

Beginning with the Yankee Rowe Nuclear Plant in 1966, more than 25 steam generator water hammers have been reported to the NRC. These water hammers or flow instabilities have involved 15 of the 34 presently operating PWRs, and resulted in four of these plants needing extensive repairs because of permanent pipe deformation and component damage. In 1973, the most damaging of these water hammer incidents occurred at Indian Point 2, resulting in the fracture and bulging of an 18-inch feedwater pipe and thermal deformation of the containment. In view of the frequency and potential severity of these flow instability events, various hardware modifications and operating procedures have been adopted by most U.S. and foreign utilities. These ad hoc approaches by U.S. industry were based almost entirely on qualitative descriptions of the phenomena, and accordingly a successful means to eliminate water hammer did not emerge.

In 1976, the NRC engaged an engineering firm (CREARE, Inc.) to undertake its own study to determine the cause and effects of past steam generator water hammers. The outcome of the program provided information which led to the development of procedural and design changes to avoid these incidents. CREARE utilized the results of their review of the available data from past incidents in the development and performance of its own test program which was to perform small scale tests (1/10) to qualitatively investigate the phenomena associated with slug formation and acceleration. From the test results obtained, CREARE was able to show that feedwater type spargers located within the steam generators with top feed and combined with a short length of pipe significantly reduced the overpressure magnitude (factor 5 to 10) vis a vis bottom drain spargers and long pipes. Thus, these results generally confirmed the results of the European tests to install J-Tubes in top-feed steam generators.

Large-scale testing has so far confirmed the efficacy of reducing water hammers by providing top discharge from the sparger and short pipe arrangement, as shown by testing at Trojan, Indian Point 2, Farley and St. Lucie in the U.S., and at Doel, Tihange 1, Ringhals 2 and Fessenheim 1 in Europe. A detailed discussion of most of the above work is published in NUREG-0291, entitled, "An Evaluation of PWR Steam Generator Water Hammer."

The staff believes that the evidence to date warrants the establishment of design guidelines for steam generators and the associated piping. Guidelines have been developed that may be used to reduce the probability of a damaging steam condensation induced water hammer, especially in the Westinghouse and Combustion Engineering PWR designs which use top-feed steam generators. The B&W once-through steam generators have not experienced damaging water hammer incidents, and therefore are not considered.

BRANCH TECHNICAL POSITION

Top-Feed Designs

In CP and OL application reviews, the staff requires the applicant to provide the following for steam generators utilizing top feed:

To eliminate or reduce possible water hammer in the feedwater system:

1. Prevent or delay water draining from the feedring following a drop in steam generator water level by means such as J-Tubes.
2. Minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest possible (less than seven feet) horizontal run of inlet piping to the steam generator feedring.
3. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feedring. Provide the procedures for these tests for approval before conducting the tests.

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SECTION 10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - None

I. AREAS OF REVIEW

At the construction permit (CP) stage, ETSB reviews the steam generator blowdown system (SGBS), as described in the applicant's safety analysis report (SAR), in the specific areas that follow. At the operating license (OL) stage, the ETSB review consists of confirming the design accepted at the CP stage.

1. ETSB reviews the SGBS design objectives in terms of expected and design flows, process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis), and process instrumentation and controls for maintaining operations within established parameter ranges.

2. ETSB reviews the seismic design and quality group classifications for equipment, piping, and components.

The liquid and gaseous waste treatment aspects of the SGBS are reviewed under SRP Sections 11.2 and 11.3. Liquid and gaseous process and effluent radiological monitoring is reviewed under SRP Section 11.5 and process sampling is reviewed under SRP Section 9.3.2.

ASB evaluates the effect of system failures to assure that safety-related equipment will not be made inoperable under SRP Section 3.6.1.

II. ACCEPTANCE CRITERIA

ETSB accepts the design of the steam generator blowdown system if the following conditions are met.

1. The SGBS is sized to accommodate the design blowdown flow needed to maintain secondary coolant chemistry for normal operation, including anticipated operational occurrences. Equipment capacities are based on design blowdown flow rates and are such that temperature limits for heat-sensitive processes are not exceeded. Instrumentation and automatic controls ensure operation within design parameters.

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2. The design of the SGBS downstream of the outer containment isolation valves is in accord with the provisions of Regulatory Guide 1.143.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from this review plan as may be appropriate for a particular case.

1. ETSB considers the pressure, temperature, flow rate, secondary coolant chemistry, main condenser water inleakage, and primary to secondary leakage to determine whether the SGBS design has included the effects of normal operation and anticipated operational occurrences (e.g., main condenser inleakage or primary to secondary leakage). ETSB determines that the design parameters are reasonable. If the proposed system includes processes which are heat-sensitive (e.g., demineralization or reverse osmosis), ETSB verifies that the design includes instrumentation and controls to protect the temperature sensitive elements. ETSB ensures that instrumentation and process controls are provided to control flashing, liquid levels, and process flow through the proper components for the radioactivity levels expected.

2. ETSB compares the quality group and seismic design classifications of SGBS components to the guidelines of Regulatory Guide 1.143.

3. ETSB reviews the waste treatment and radiological process and effluent monitoring aspects of the SGBS in SRPs 11.2, 11.3, 11.5 and 9.3.2. ETSB reviews the proposed piping and instrumentation diagrams (P&IDs) and process flow diagrams, the method of operation, the processing to be provided, and the interfaces between the blowdown system and other plant systems to determine: (1) whether unusual design conditions exist which could lead to safety problems, and (2) that the system is capable of performing its intended functions.

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 steam generator blowdown system (SGBS) controls the concentration of chemical impurities and radioactive materials in the secondary coolant. The scope of review of the SGBS included piping and instrumentation diagrams, seismic and quality group classifications, design process parameters, and instrumentation and process controls. The review has included the applicant's evaluation of the proposed system operation and the applicant's estimate of the controlling process parameters.

"We have reviewed the capability of the system to process blowdown flows and to maintain secondary coolant chemical and radiological limits, and the capability of the instrumentation and process controls to maintain system operation within the proposed limits.

"The basis for acceptance in our review has been conformance of the applicant's designs and design criteria to applicable regulatory guides.

"Based on the foregoing evaluation, we conclude that the proposed steam generator blowdown system is acceptable."

V. REFERENCES

1. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components in Light-Water-Cooled Nuclear Reactor Power Plants."

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SECTION 10.4.9 AUXILIARY FEEDWATER SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Reactor Systems Branch (RSB)
Instrumentation and Control Systems Branch (ICSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The auxiliary feedwater system (AFS) normally operates during startup, hot standby and shutdown as the feedwater system for pressurized water reactor (PWR) plants. In conjunction with a seismic Category I water source, it also functions as an emergency system for the removal of heat from the primary system when the main feedwater system is not available for emergency conditions including small LOCA cases. The AFS operates over a time period sufficient either to hold the plant at hot standby for several hours or to cool down the primary system, at a rate not to exceed limits specified in technical specifications, to temperature and pressure levels at which the low pressure decay heat removal system can operate. The design of the AFS should meet the requirements of General Design Criteria 2, 4, 5, 19, 44, 45 and 46.

The ASB reviews the AFS from the condensate storage tank (normal operation), or the seismic Category I water supply including valving and cross-connects (emergency operation), to the connections with the steam generators, which are made either through a connection to the main feedwater piping or through separate auxiliary feedwater piping connected directly to the steam generators. All inter-connections and cross-connections are included in the review.

The review also includes AFS components, e.g., pumps, valves, and piping, with respect to their functional performance as affected by adverse environmental occurrences, abnormal operational requirements, and off-normal conditions, e.g., small breaks in the primary system or the loss of offsite power.

The system is reviewed to determine that a single malfunction, a failure of a component, or the loss of a cooling source does not reduce the safety-related functional performance capabilities of the system. The ASB review assures that:

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1. System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from the effects of internally and externally generated missiles.

2. The system satisfies the recommendations of Branch Technical Position ASB 3-1 with respect to the effects of pipe whip and jet impingement that may result from high or moderate energy piping breaks or cracks (in this regard the AFS is considered to be a high energy system).

3. The system and components satisfy design code requirements, as appropriate for the assigned quality group and seismic classifications.

4. The failure of non-essential equipment or components does not affect essential functions of the system.

5. The system is capable of withstanding a single active failure.

6. The system possesses diversity in motive power sources such that system performance requirements may be met with either of the assigned power sources, e.g., a system with an a-c subsystem and a redundant steam/d-c subsystem.

7. The system design precludes the occurrence of fluid flow instabilities, e.g., water hammer, in system inlet piping during normal plant operation or during upset or accident conditions (see SRP Section 10.4.7).

8. Functional capability is assured by suitable protection during abnormally high water levels (adequate flood protection considering the probable maximum flood).

9. The capability exists to detect, collect, and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.

10. Provisions are made for operational testing.

11. Instrumentation and control features are provided to verify the system is operating in a correct mode.

12. The system is capable of automatically initiating auxiliary feedwater flow upon receipt of a system actuation signal.

13. The system satisfies the recommendations of Regulatory Guide 1.62 with respect to the system capability to manually initiate protective action by the auxiliary feedwater system.

14. The system design possesses the capability to automatically terminate auxiliary feedwater flow to a depressurized steam generator, and to automatically provide feedwater to the intact steam generator.

15. The system possesses sufficient auxiliary feedwater flow capacity so that a cold shutdown can be achieved.

16. The applicant's proposed technical specifications are such as to assure the continued reliability of the AFS during plant operation; i.e., the limiting conditions for operation and the surveillance testing requirements are specified and are consistent with those for other similar plants.

Secondary review evaluations are performed by other branches and the results used by the ASB to complete the overall evaluation of the system. The secondary reviews are as follows. The RSB identifies any functional interfaces between essential components of the reactor coolant or emergency core cooling systems and the AFS that are required for operation during normal operations or accident conditions. The RSB establishes post-accident heat loads and the associated time intervals available for cooling various components. The SEB determines the acceptability of the design analyses, procedures and criteria used for seismic Category I structures that must withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification testing and operability of components and confirms that components, piping, and structures 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 evaluate system controls, instrumentation, and power sources with respect to capability, capacity, and reliability during normal and emergency conditions.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the auxiliary feedwater system, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. Listed below are the specific criteria as they relate to the AFS.

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, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. General Design Criterion 19, as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.

5. General Design Criterion 44, to assure:

   a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

   b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)

   c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

6. General Design Criterion 45, as related to design provisions made to permit periodic inservice inspection of system components and equipment.

7. General Design Criterion 46, as related to design provisions made to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

8. Regulatory Guide 1.26, as related to the quality group classification of system components.

9. Regulatory Guide 1.29, as related to the seismic design classification of system components.

10. Regulatory Guide 1.62, as related to design provisions made for manual initiation of each protective action.

11. Regulatory Guide 1.102, as related to the protection of structures, systems, and components important to safety from the effects of flooding.

12. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles.

13. Branch Technical Positions ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

14. Branch Technical Position ASB 10-1, as related to auxiliary feedwater pump drive and power supply diversity.

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 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) 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 procedures for OL applications also include 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 as a result of the staff's review.

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.

For the purpose of this SRP section, a typical system is assumed which has redundant auxiliary feedwater trains, with a 50% capacity motor-driven pump in each train feeding directly to the steam generators, and a 100% capacity steam turbine-driven pump able to supply either of the redundant trains. The pumping capacity should permit the system to hold the plant at hot standby and subsequently to cool down the reactor at specified cooldown rates. This requirement should also be met for conditions involving a small break area loss-of-coolant accident (LOCA) or a pipe break outside containment. For cases where there are variations from the typical arrangement, the reviewer adjusts the review procedures to suit the design. However, the system design is required to meet the acceptance criteria given in subsection II.

1. The SAR is reviewed to determine that the system description and piping and instrumentation diagrams (P&IDs) identify the AFS equipment and arrangement that is used for normal operation and for safe plant shutdown (essential) operation. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed to verify that:

   a. Minimum performance requirements for the system are sufficient for the various functions of the AFS.

   b. Essential portions of the AFS are isolable from non-essential portions, so that system performance is not impaired in the event of a failure of a non-essential component.

c. Component and system descriptions in the SAR include appropriate seismic and quality group classifications, and the P&IDs indicate any points of change in piping quality group classification.

d. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary recirculation loops around pumps or isolation valves as may be required by this program.

2. The reviewer verifies that the system safety function will be maintained as required, in the event of adverse environmental phenomena, breaks or cracks in fluid system piping outside containment, system component failures, loss of an onsite motive power source, or loss of offsite power. The reviewer uses engineering judgment and the results of failure modes and effects analyses to determine that:

a. The failure of portions of the system or of other systems not designed to seismic Category I standards and located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the AFS, will not preclude operation of the essential portions of the AFS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems.

b. The essential portions of the AFS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the SRP Section 3 series. The location and design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or the components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. The essential portions of the system are protected from the effects of high and moderate energy line breaks in accordance with Branch Technical Position ASB 3-1. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the AFS, or that protection from the effects of failure will be provided. The means of providing such protection will generally be given in Section 3.6 of the SAR and procedures for reviewing this information are given in SRP Section 3.6.

d. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The SAR is reviewed to see that for each AFS component or subsystem affected by the loss of offsite power, system flow and heat transfer capability meet minimum requirements. Statements in the SAR and the results of failure modes and effects analyses are considered in assuring that the system meets these requirements.

e. The system is designed with adequate redundancy to accommodate a single active component failure without loss of function.

f. Diversity in pump motive power sources and essential instrumentation and control power sources has been provided. The diverse system including pump(s), controls and valves should be independent of offsite and onsite AC power sources in accordance with the guidelines of Branch Technical Position ASB 10-1.

g. The system is designed with adequate instrumentation to automatically initiate auxiliary feedwater flow to the steam generators upon receipt of an actuation signal. The initiation signal should start all auxiliary feedwater pumps and supporting systems, align the auxiliary feedwater sources, and open flow paths from the auxiliary feedwater pumps to the steam generator(s).

h. The system is designed with the capability to manually initiate the protective actions necessary so that the auxiliary feedwater system design satisfies the recommendation of Regulatory Guide 1.62.

i. The AFS is designed with redundant instrumentation so that the system will automatically limit or terminate auxiliary feedwater flow to a depressurized steam generator, and to assure that the minimum required flow is directed to the intact steam generator(s).

j. The AFS is designed with sufficient flow capacity so that the system can remove residual heat over the entire range of reactor operation and achieve a cold shutdown condition.

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 auxiliary feedwater system includes all components and equipment from the condensate storage tank (normal operation) or the seismic Category I emergency water supply (including valves and cross connections) to the connection with the steam generators. Based on the review of the applicant's proposed design criteria, design bases and safety classification for the auxiliary feedwater system, and

system performance requirements during normal, abnormal, and accident conditions, the staff concludes that the design of the auxiliary feedwater system and supporting systems 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 5, "Sharing of Structures, Systems, and Components," General Design Criterion 19, "Control Room," General Design Criterion 44, "Cooling Water," General Design Criterion 45, "Inspection of Cooling Water System," and General Design Criterion 46, "Testing of Cooling Water System," and meets the guidelines contained in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.29, "Seismic Design Classification," Regulatory Guide 1.62, "Manual Initiation of Protective Actions," Regulatory Guide 1.117, "Tornado Design Classification," Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWRs," and Branch Technical Position ASB 3-1 and MEB 3-1, 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 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

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. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

9. Regulatory Guide 1.29, "Seismic Design Classification."

10. Regulatory Guide 1.62, "Manual Initiation of Protective Actions."

11. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

12. Regulatory Guide 1.117, "Tornado Design Classification."

13. 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.

14. Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

BRANCH TECHNICAL POSITION ASB 10-1

DESIGN GUIDELINES FOR AUXILIARY FEEDWATER SYSTEM PUMP DRIVE AND POWER SUPPLY DIVERSITY FOR PRESSURIZED WATER REACTOR PLANTS

A. BACKGROUND

Heat removal from pressurized water reactor plants following reactor trip and a loss of offsite power is accomplished by the operation of several systems including the secondary system via the steam relief system. Similar capability is required to mitigate the consequences of certain postulated piping breaks. Such heat removal involves heat transfer from the reactor to the steam generators, resulting in the production of steam which is then released to the atmosphere. In this process it becomes necessary to supply makeup water to the steam generators. This is accomplished by the use of an auxiliary feedwater system, which generally consists of redundant components that are powered by both electrical and steam-driven sources.

The auxiliary feedwater system functions as an engineered safety system because it is the only source of makeup water to the steam generators for decay heat removal when the main feedwater system becomes inoperable. It must, therefore, be designed to operate when needed, using the principles of redundancy and diversity in order to assure that it can function under postulated accident conditions. The majority of current systems are powered by electrical or steam-driven sources. Operating experience demonstrates that each type of motive power can be subject to a failure of the driving component itself, its source of energy, or the associated control system. The effects of such failures can be minimized by the utilization of diverse systems that include energy sources of at least two different and distinct types.

The provision of several independent flow paths for the auxiliary feedwater system serves to preclude the possibility of a complete loss of function due to a single event, either occurring alone, or in conjunction with the failure of an active component. The auxiliary feedwater system is categorized as a high energy system, because either that section of line which connects to the main feedwater piping or the steam generator is pressurized during plant operation or else the entire system is pressurized when in use during startup, hot standby, and shutdown.

The staff believes that it is necessary to establish design guidelines for the auxiliary feedwater system, and in this regard has developed guidelines that may be used to select the minimum diversity acceptable for auxiliary feedwater system pump drives and power supplies.

B. BRANCH TECHNICAL POSITION

1. The auxiliary feedwater system should consist of at least two full-capacity, independent systems that include diverse power sources.

2. Other powered components of the auxiliary feedwater system should also use the concept of separate and multiple sources of motive energy. An example of the required diversity would be two separate auxiliary feedwater trains, each capable of removing the afterheat load of the reactor system, having one separate train powered from either of two a-c sources and the other train wholly powered by steam and d-c electric power.

3. The piping arrangement, both intake and discharge, for each train should be designed to permit the pumps to supply feedwater to any combination of steam generators. This arrangement should take into account pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One arrangement that would be acceptable is crossover piping containing valves that can be operated by remote manual control from the control room, using the power diversity principle for the valve operators and actuation systems.

4. The auxiliary feedwater system should be designed with suitable redundancy to offset the consequences of any single active component failure; however, each train need not contain redundant active components.

5. When considering a high energy line break, the system should be so arranged as to assure the capability to supply necessary emergency feedwater to the steam generators, despite the postulated rupture of any high energy section of the system, assuming a concurrent single active failure.

C. REFERENCES

None

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SECTION 11.1 SOURCE TERMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - None

I. AREAS OF REVIEW

ETSB reviews the sources of radioactivity that are input to the radioactive waste management systems employed for treatment of liquid, gaseous and solid wastes.

1. Review of radioactive source terms includes consideration of parameters used to determine the concentration of each isotope in the reactor coolant; fraction of fission product activity released to the reactor coolant; concentrations of all nonfission product radioactive isotopes in the reactor coolant; leakage rates and associated fluid activity for all potentially radioactive water and steam systems; and potential sources of radioactive materials in effluents that are not considered in the applicant's safety analysis report (SAR) Section 11.2, "Liquid Waste Management Systems," and SAR Section 11.3, "Gaseous Waste Management Systems." The following release points are considered in the evaluation of effluent releases:

   a. Boiling water reactor (BWR) gaseous wastes (noble gases, radioiodine, and particulates carbon-14 and tritium), consisting of offgases from the main condenser evacuation and turbine gland sealing systems, steam and liquid leakage to containment, radwaste, turbine, fuel handling and auxiliary buildings, and ventilation system exhausts from buildings having the potential for containing radioactive materials.

   b. BWR liquid wastes, consisting of leakage to equipment and floor drains from buildings housing equipment and components that may contain radioactive fluids; contaminated liquids produced by plant operations, such as demineralizer regenerants and resin sluice water, filter backwashes, ultrasonic resin cleaning rinses, decontamination solutions, and laboratory samples and rinses; and detergent wastes.

   c. Pressurized water reactor (PWR) gaseous wastes (noble gases, radioiodine, and particulates carbon-14 and tritium), consisting of offgases from the steam generator blowdown flash tank; offgases from the main condenser evacuation system; leakage to containment, fuel handling auxiliary, and turbine buildings; noble gases stripped from the primary coolant during normal operation and at

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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.

shutdown; and cover and vent gases from tanks and equipment containing radioactive material.

d. PWR liquid wastes, consisting of primary coolant processed to control boron concentration (shim bleed); leakage collected in equipment and floor drains from buildings housing equipment and components that may contain radioactive fluids; steam generator blowdown; condensate demineralizer regenerant solutions; contaminated liquids from anticipated plant operations such as resin sluices, filter backwashes, decontamination solutions, and sample station drains; and detergent wastes.

2. The calculated releases of radioactive materials in liquid and gaseous effluents will be used in SRP Sections 11.2 and 11.3 to evaluate the liquid and gaseous waste systems.

II. ACCEPTANCE CRITERIA

ETSB will accept the source terms used as the design basis for expected releases if the following criteria are met:

1. The parameters used to calculate primary and secondary (PWR) coolant concentrations are consistent with those given in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," and NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors."

2. All normal and potential release points of radioactive effluent delineated in subsection I are considered.

3. For each source of liquid and gaseous waste considered in subsection II.1.b, the volumes and radioactivity levels given for normal operation and anticipated operational occurrences are consistent with those given in NUREG-0016 and NUREG-0017.

4. Reduction factors for special design features used to reduce leakage, such as clean sealing steam for valve stems and turbine glands, are consistent with those given in NUREG-0016 and NUREG-0017.

5. Decontamination factors for inplant control measures used to reduce releases to the environment, such as iodine removal systems and high efficiency particulate air (HEPA) filters for building ventilation exhaust systems, are consistent with those given in NUREG-0016 and NUREG-0017.

An acceptable method for satisfying the criteria of subsection II.1 consists of using the gaseous and liquid effluent (GALE) Computer Code and the source term parameters given in NUREG-0016 and NUREG-0017. A complete Fortran listing of the PWR and BWR GALE computer code is given in these reports. The applicant should provide the information listed in Appendix A (BWRs) and Appendix B (PWRs) to Regulatory Guide 1.112 (Ref. 5).

If the applicant's calculational technique or any source term parameter differs from that given in NUREG-0016 and NUREG-0017, they should be described in detail and the bases for the methods and parameters used should be provided.

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 review of the mathematical models and parameters given in the SAR to calculate primary coolant concentrations, and of the leakage rates to the radioactive waste management systems, ETSB compares parameters and calculations given in the SAR with the models and parameters given in NUREG-0016 and NUREG-0017. If the SAR includes models or parameters to estimate primary coolant concentrations and leakage rates that differ from these reports, the parameters and calculations used need to be substantiated. The preferred method of substantiation is by presentation of operating data from similar reactors.

2. ETSB performs an independent calculation of the primary and secondary (PWR) coolant concentrations and of the release rates of radioactive materials using the GALE Computer Code, and the "Principal Parameters for Source Term Calculations" given in NUREG-0016 and 0017, and the information supplied by the applicant in accordance with Appendices A and B of Regulatory Guide 1.112.

3. In the calculation, ETSB will use the applicant's values as given in the SAR for the following parameters: design core thermal power level, steam flow rate, coolant mass, and coolant purification rates.

4. ETSB will use the primary coolant concentrations and leakage rates calculated above as inputs for evaluation of the liquid waste system, under SRP Section 11.2, and the gaseous waste systems, under SRP Section 11.3, to determine if the radioactive waste management systems meet the dose design objectives of Appendix I to 10 CFR Part 50.

5. The ETSB source term calculations are used for both the review of the SAR and for the staff's Environmental Impact Statement.

IV. EVALUATION FINDINGS

The ETSB summary statement on the acceptability of source terms used as design parameters for the waste management systems will be made under SRP Section 11.2, "Liquid Waste Systems," and 11.3, "Gaseous Waste Systems."

V. REFERENCES

1. 10 CFR Part 20 "Standards for Protection Against Radiation."

2. 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," May 5, 1975.

3. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)."

4. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs).

5. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluent from Light-Water-Cooled Power Reactors."

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SECTION 11.2 LIQUID WASTE MANAGEMENT SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Radiological Assessment Branch (RAB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

At the construction permit (CP) stage, ETSB reviews the information in the applicant's preliminary safety analysis report (PSAR) in the specific areas that follow. During the operating license (OL) stage of review, 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 liquid radwaste treatment system design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The ETSB review will include the system piping and instrumentation diagrams (P&IDs), and process flow diagrams showing methods of operation and factors that influence waste treatment, e.g., system interfaces and potential bypass routes.

2. Equipment design capacities, expected flow and radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup time. The system design capacity relative to the design and expected input flows, and the period of time the system is required to be in service to process normal waste flows. The availability of standby equipment, alternate processing routes, and interconnections between subsystems. This information is used in the ETSB review to evaluate the overall system capability to meet anticipated demands imposed by major processing equipment downtime and waste volume surges due to anticipated operational occurrences.

3. The quality group classifications of piping, and equipment, and the bases governing the design criteria chosen. Provisions to prevent, control and collect releases of radioactive material in liquids due to tank overflows from all plant systems, outside reactor containment having the potential to incur such releases. Design and expected temperatures and pressures, and materials of construction of the components of the liquid waste management system.

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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. Design provisions incorporated in the equipment and facility design to reduce leakage and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143. (Ref. 10)

5. Special design features that would reduce liquid input volumes or discharge of radioactive material in liquid effluents. Special design features, topical reports incorporated by reference, and data obtained from previous experience with similar systems which are submitted with the SAR.

6. The technical specifications proposed by the applicant for process and effluent control will be reviewed at the operating license stage (FSAR).

Design provisions incorporated to sample and monitor radioactive materials in liquid process and effluent streams are reviewed under SRP Sections 11.5 and 9.3.2.

RAB will provide calculated doses based on the ETSB liquid source terms for inclusion in the staff's Environmental Impact Statement and Safety Evaluation Report.

SEB evaluates the applicant's proposed seismic design of structures housing the liquid radwaste system for inclusion in the staff's Safety Evaluation Report.

The consequences of liquid tank failures having the potential to release radioactive liquids are evaluated in SRP Section 15.7.3.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. The liquid radwaste treatment system should have the capability to meet the requirements specified in 10 CFR Parts 20 and 50 and the dose design objectives specified in Appendix I to 10 CFR Part 50, including provisions to treat liquid radioactive waste such that:

   a. The calculated annual total quantity of all radioactive material released from each reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

   b. In addition to a. above, the liquid radwaste treatment systems should include all items of reasonably demonstrated technology that when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor.

c. The concentrations of radioactive materials in liquid effluents released to an unrestricted area should not exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

2. The liquid radwaste treatment system should be designed to meet the anticipated processing requirements of the station. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. ETSB will accept systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences. To meet these processing demands, ETSB will consider interconnections between subsystems, redundant equipment, and reserve storage capacity.

3. The seismic design of structures housing liquid radwaste systems, the quality group classification of liquid radwaste treatment equipment, and provisions to prevent and collect spills from indoor and outdoor storage tanks should conform to the guidelines of Regulatory Guide 1.143.

4. ETSB will accept system designs that contain provisions to control leakage and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143.

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 of the liquid waste treatment system, the P&IDs and system process flow diagrams are reviewed to determine all sources of liquid input volumes, the points of collection of liquid waste, the flow paths of liquids through the system including all bypasses, the treatment provided, and the points of release of liquid effluents to the environment. This information is used to calculate the quantity of radioactive materials released annually in liquid effluents during normal operation, including anticipated operational occurrences, using the parameters given, the GALE Code, and calculational techniques given in NUREG-0016 and NUREG-0017. A complete Fortran listing of the GALE computer code is given in these reports. The results of this calculation will be used to determine whether the proposed treatment system design meets the acceptance criterion of II.1.c. Compliance with the acceptance criteria given in subsection II.1.a concerning exposures to the total body or critical organ of an individual in an unrestricted area will be determined based on RAB dose calculations using the ETSB-calculated source term.

Compliance with the acceptance criterion given in II.1.b concerning the cost-benefit analysis will be determined based on RAB man-rem dose calculations in conjunction with ETSB cost-benefit studies.

2. The ETSB review of the liquid waste treatment system design capacity will encompass three major areas:

   a. The system capability to process wastes in the event of a single major equipment item failure, e.g., an evaporator outage.

   b. The system capability to accept additional wastes during operations which result in excessive liquid waste generation.

   c. The system capability to process wastes at design basis fission product leakage levels, i.e., from 1% of the fuel producing power in a PWR or, in a BWR, consistent with a noble gas release of 100 μCi/sec/MWt measured after 30 minutes delay.

   ETSB will compare the average input flows to the design flows to determine the fraction of time individual subsystems must be online to process normal waste inputs. ETSB will review the operational flexibility designed into the system, i.e., cross connections between subsystems, redundant or reserve processing equipment, and reserve storage capacity. Based on the usage factors and operational flexibilities, ETSB will evaluate the overall system capability to process wastes in the event of (a), (b), or (c), above, by comparing the design flows to the potential process routes and equipment capacities. ETSB will assume evaporators are unavailable for 2 consecutive days per week for maintenance. If two days holdup capacity or an alternative evaporator are not available for the process stream, ETSB will assume the stream is processed by an alternate route or discharged to the environment, consistent with the guidelines of NUREG-0016 and NUREG-0017.

3. ETSB compares the quality group classification for radwaste systems with the guidelines of Regulatory Guide 1.143. ETSB assures that the design includes provisions to prevent and collect leakage due to overflows and spillage from indoor and outdoor storage tanks, are in conformance with the guidelines of Regulatory Guide 1.143. SEB reviews the seismic design criteria of structures housing the liquid radwaste system in accordance with the design guidance identified in Regulatory Guide 1.143.

4. ETSB compares the system design, system and building layout, equipment design, method of operation, and provisions to reduce leakage and facilitate operations and maintenance with the guidelines of Regulatory Guide 1.143. ETSB will evaluate special design features provided to control leakage from system components and topical reports on systems designed on a case-by-case basis.

5. ETSB reviews the technical specifications proposed by the applicant for process and effluent control. The reviewer will determine that the content and intent of the technical specifications are in agreement with the requirements developed as a result of the staff's review. The review will include the evaluation or

development of appropriate limiting conditions for operation and their bases consistent with the plant design.

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 liquid radwaste treatment systems include the equipment and instrumentation to control the release of radioactive materials in liquid effluents."

In our evaluation, we have considered releases of radioactive materials in liquid effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant for each reactor on the _____ site. We have determined that the proposed liquid radwaste treatment systems are capable of maintaining releases of radioactive materials in liquid effluents such that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 3 millirems to the total body and 10 millirems to any organ.

We have also considered the potential effectiveness of augmenting the proposed liquid radwaste treatment systems using items of reasonably demonstrated technology and have determined that further effluent treatment will not effect reductions in the cumulative population dose reasonably expected within a 50-mile radius of the reactor at a cost of less than $1000 per man rem or man-thyroid-rem.

We have also considered the potential consequences resulting from reactor operation, and we have determined the concentrations of radioactive materials in liquid effluents in unrestricted areas will be a small fraction of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

We have considered the capabilities of the proposed liquid radwaste treatment system to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.

We have reviewed the applicant's quality assurance provisions for the liquid radwaste systems, the quality group classifications used for system components, and the seismic design applied to structures housing these systems. The design of the systems and structures housing these systems meet the criteria as set forth in Regulatory Guide 1.143.

We have reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in liquids due to inadvertent tank overflows and conclude that the measures proposed by the applicant are consistent with the criteria as set forth in Regulatory Guide 1.143.

Based on the foregoing evaluation, we conclude that the proposed liquid radwaste treatment system is acceptable. The basis for acceptance has been conformance of the applicant's design, design criteria, and design bases for the liquid radioactive waste treatment systems to the Commission's regulations and to applicable Regulatory Guides, as referenced above.

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation," and Appendix B, "Concentration in Air and Water Above Natural Background."

2. 10 CFR § 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents - Nuclear Power Reactors."

3. 10 CFR § 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection."

6. 10 CFR Part 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.

7. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

8. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)."

9. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs)."

10. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components in Light-Water-Cooled Nuclear Reactor Power Plants."

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SECTION 11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Radiological Assessment Branch (RAB)
- Accident Analysis Branch (AAB)
- Structural Engineering Branch (SEB)

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, 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 gaseous radwaste treatment and ventilation system designs, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials (noble gases, radioiodine, and particulates) in gaseous effluents. The ETSB review will include the system piping and instrumentation diagrams (P&IDs), and the process flow diagrams showing methods of operation and factors that influence waste treatment, e.g., system interfaces and potential bypass routes.

2. Equipment and ventilation system design capacities, expected flows and radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup time. The system design capacity relative to the design and expected input flows, the period of time the system is required to be in service to process normal waste flows, availability of standby equipment, alternate processing routes, and interconnections between subsystems. This information is used to evaluate the overall system capability to meet anticipated demands imposed by major processing equipment downtime and waste volume surges due to anticipated operational occurrences.

3. The quality group classifications of piping and equipment, and the bases governing the design criteria chosen. Design and expected temperatures and pressures, and materials of construction of the components of the system.

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.

4. Design provisions incorporated in the equipment and facility design to facilitate operation and maintenance in conformance with the guidelines of Regulatory Guide 1.143. (Ref. 9)

5. Special design features to reduce leakage of gaseous waste or discharge of radioactive material in gaseous effluents. Special design features, topical reports incorporated by reference, and data obtained from previous experience with similar systems which are submitted with the SAR.

6. Design features to preclude the possibility of an explosion if the potential for explosive mixtures exist.

7. The technical specifications proposed by the applicant for process and effluent control are reviewed at the operating license stage (FSAR).

Design provisions incorporated to sample and monitor radioactive materials in gaseous process and effluent streams are reviewed under SRP Section 11.5.

RAB provides calculated doses based on the ETSB gaseous source term for inclusion in the staff's Environmental Impact Statement and Safety Evaluation Report.

SEB will provide an evaluation of the applicant's proposed seismic design of the gaseous waste processing system and of structures housing this system.

AAB will calculate the doses that result as the consequence of a gas decay tank rupture under SRP 15.7.1 when requested by ETSB.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. The gaseous radwaste treatment system should have the capability to meet the requirements of 10 CFR Part 20 and the dose design objectives specified in Appendix I to 10 CFR Part 50, including provisions to treat gaseous radioactive wastes such that:

   a. The calculated annual total quantity of all radioactive material released from each reactor at the site to the atmosphere will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

   b. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form released from each reactor at the site in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in

particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

c. In addition to a. and b., above, the gaseous radwaste treatment systems should include all items of reasonably demonstrated technology that when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor.

d. The concentrations of radioactive materials in gaseous effluents released to an unrestricted area should not exceed the limits in 10 CFR Part 20.

2. The gaseous radwaste treatment system should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. ETSB will accept systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences. To meet these processing demands, ETSB will consider shared systems, redundant equipment, and reserve storage capacity.

3. The seismic design and quality group classification of components used in the gaseous waste treatment system and structures housing this system should conform to the guidelines of Regulatory Guide 1.143. The design should include precautions to stop continuous leakage paths, i.e., to provide liquid seals downstream of rupture discs and to prevent permanent loss of the liquid seals in the event of an explosion.

4. ETSB will accept system designs that contain provisions to control leakage and to facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143 (Ref. 10).

5. ETSB will use the guidelines in Regulatory Guide 1.140 (Ref. (11) for the design, testing and maintenance of HEPA filters and charcoal adsorbers installed in normal ventilation exhaust systems.

   If decontamination factors for iodine different from those specified in Regulatory Guide 1.140 are used for design purposes, they should be supported by test data under operating or simulated operating conditions (temperature, pressure, humidity, expected iodine concentrations, and flow rate). The effects of aging and poisoning by airborne contaminants should also be supported by test data.

6. If the potential for an explosive mixture of hydrogen and oxygen exists, the gaseous waste handling and treatment system should either be designed to withstand

the effects of a hydrogen explosion, or be provided with dual gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures.

a. For a system designed to withstand the effects of a hydrogen explosion, the design pressure of the system should be approximately 20 times the operating absolute pressure (including the intermediate stage condenser for BWR offgas systems).

   Small allowances should be made to conform to standard design pressures for off-the-shelf components; i.e., if the system operating pressure is nominally 15 psia but could approach 20 psia by design, piping could be designed to 350 psia, since the next higher standard pressure rating is 600 psia.

   The process gas stream should be analyzed for potentially-explosive mixtures and annunciated both locally and in the control room.

b. For systems not designed to withstand a hydrogen explosion, dual gas analyzers (with dual being defined as two independent gas analyzers continuously operating and providing two independent measurements verifying that hydrogen and/or oxygen are not present in potentially-explosive concentrations) with automatic control functions are required to preclude the formation or buildup of explosive hydrogen/oxygen mixtures. Gas analyzers should annunciate alarms both locally and in the control room. "High alarm" should be set approximately 2% and "High-high alarm" should be set at a maximum of 4% hydrogen or oxygen.

Control features to reduce potential for explosion should be automatically initiated at "High-high alarm" setting. The automatic control features should be as follows: (1) for systems designed to preclude explosions by maintaining either hydrogen or oxygen below 4%, the source of hydrogen or oxygen (as appropriate) should be automatically isolated from the system (valve should fail in closed position); (2) for systems using recombiners, if the downstream hydrogen and/or oxygen concentration exceeds 4% (as appropriate), acceptable control features include automatically switching to an alternate recombiner train; and (3) injection of diluents to reduce concentrations below the limits specified herein.

Systems designed to operate below 4% hydrogen and below 4% oxygen may be analyzed for either hydrogen or oxygen; systems designed to operate below 4% hydrogen only (no oxygen restrictions), should be analyzed for hydrogen; and systems designed to operate above 4% hydrogen, should be analyzed for oxygen.

For BWR systems with steam dilution upstream of the recombiners, analysis for hydrogen (oxygen is not an acceptable alternative) should be downstream of the recombiners and upstream of the delay portions of the system (analysis upstream of the recombiners is not required if the system is designed to assure the availability of dilution steam during operation). For PWR systems using recombiners, analysis for hydrogen and/or oxygen should be downstream of the recombiners. In addition, unless the system design features preclude explosive mixtures of hydrogen and oxygen upstream of the

recombiners, analysis for hydrogen and/or oxygen (as appropriate) should be upstream of the recombiners as well. The number of gas analyzers and control features at each location should be in accordance with this SRP. One gas analyzer upstream and one gas analyzer downstream of the recombiners should not be construed as dual gas analyzers. For systems involving pressurized storage tanks (excluding surge tanks), at least one gas analyzer is required between the compressor and the storage tanks. Dual gas analyzers set to sequentially measure concentrations both upstream and downstream of a recombiner are acceptable for a PWR. When two or more potentially explosive process streams are combined before entering a component, each stream or the combination thereof, is required to have dual gas analyzers.

If gas analyzers are to be used to sequentially measure several points in a system not designed to withstand a hydrogen explosion, at least one gas analyzer which is continuously on stream is required. The continuous gas analyzer should be at a point common to streams measured sequentially; i.e., should be sampling the combined stream.

Gas analyzers should have daily sensor checks, monthly functional checks and quarterly calibrations.

Gas analyzers installed in systems designed to withstand a hydrogen explosion should also be capable of withstanding a hydrogen explosion; gas analyzers installed in the systems not designed to withstand a hydrogen explosion need not be capable of withstanding a hydrogen explosion (similar requirements apply to radiation monitors which are internal to lines containing potentially explosive mixtures).

All gas analyzers shall be nonsparking.

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 of the gaseous waste treatment system, the P&IDs and system process flow diagrams are reviewed to determine all sources of gaseous waste, the points of collection of gaseous wastes, the flow paths of gases through the systems, including all bypasses, the treatment provided and the points of release of gaseous effluents to the environment. This information is used to calculate the quantity of radioactive material (noble gases, radioiodine, and particulates) released annually in gaseous effluents during normal operations, including anticipated operational occurrences, using the given parameters, the GALE Code, and the calculational techniques given in NUREG Reports 0016 and 0017. A complete Fortran listing of the GALE computer code is given in these reports. The results of this calculation will be used to determine whether the proposed treatment system design meets the acceptance criterion of II.1.d. Compliance with the acceptance criteria of II.1.a, and b, concerning exposures of the total body, skin, and thyroid will be determined based on RAB dose calculations using the ETSB calculated source term. Conformance with the acceptance criterion given in II.1.c concerning the cost-benefit analysis will be

determined based on RAB man-rem dose calculations in conjunction with ETSB cost-benefit studies.

2. The ETSB review of the gaseous waste treatment system design capacity will encompass two major areas:

   a. The capability of the system to process gaseous wastes in the event of a single major equipment item failure. For nonredundant equipment or components, ETSB will assume a 3-week downtime every other year (10 days per year average).

   b. The capability of the system to process gaseous wastes at design basis fission product levels, i.e., from 1% of the fuel producing power in a PWR or, in a BWR, consistent with a noble gas release rate of 100 μCi/sec/MWt at 30 minutes delay.

   ETSB will review the operational flexibilities designed into the system, e.g., cross connections between subsystems, redundant or reserve processing equipment, and reserve storage capacity.

   In the evaluation of charcoal delay systems for radioactive gas decay, ETSB considers the bed dimensions, mass of charcoal, flow rate, temperatures, pressures, humidity, and dynamic adsorption coefficients to calculate the effective holdup times.

3. ETSB compares the quality group classification of the gaseous waste system, and the seismic design of both gaseous waste systems and the structures housing these systems with the guidelines of Regulatory Guide 1.143. SEB reviews the seismic design criteria of equipment and of structures housing the gaseous radwaste treatment systems in accordance with the design guidance identified in Regulatory Guide 1.143. ETSB also determines if the applicant's design includes adequate provisions to stop continuous leakage paths after an explosion. The areas of concern are (1) streams where water decomposition gases (hydrogen and oxygen) exist in a BWR, (2) cover gas streams where air inleakage can occur in a PWR, and (3) where there is a possibility of liquid hydrocarbons and ozone collecting in a cryogenic distillation system.

4. ETSB will compare the system design, system layout, equipment design, method of operation, and provisions to reduce leakage and to facilitate operations and maintenance to the guidelines of Regulatory Guide 1.143. ETSB will evaluate special design features provided to control leakage from system components and topical reports on system designs on a case-by-case basis.

5. ETSB will compare the design testing and maintenance criteria for HEPA filters and charcoal adsorbers in filtration systems with the guidelines of Regulatory Guide 1.140.

6. If there is a potential that explosive hydrogen/oxygen mixtures exist, ETSB will determine, using the system description and P&IDs, whether the applicant has designed the gaseous radwaste treatment system to withstand the effects of such an explosion,

or has provided the required dual instrumentation to annunciate and prevent the buildup of potentially explosive mixtures.

7. At the OL stage ETSB will review the technical specifications proposed by the applicant for process and effluent control. The reviewer will determine that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements developed as a result of the staff's review. The review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

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 gaseous radwaste treatment system includes the equipment and instruments to control the release of radioactive materials in gaseous effluents."

In our evaluation, we have considered radionuclide releases of radioactive material (noble gases, radioiodine and particulates) in gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant for each reactor on the (_____) site. We have determined that the proposed gaseous radwaste treatment systems are capable of maintaining releases of radioactive materials in gaseous effluents such that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 5 mrem to the total body and 15 mrem to any organ and that releases of radioiodine and radioactive material in particulate form are less than 15 mrem to any organ.

We have also considered the potential effectiveness of augmenting the proposed gaseous radwaste treatment systems using items of reasonably demonstrated technology and have determined that "further effluent treatment will not effect reductions in the cumulative population dose within a 50-mile radius of the reactor at a cost of less than $1,000 per man-rem or $1,000 per man-thyroid-rem."

We have also considered the potential consequences resulting from reactor operation with "1% of the operating fission product inventory in the core being released to the primary coolant" for a PWR or "a fission product release rate consistent with a noble gas release rate to the reactor coolant of 100 μCi/MWt-sec at 30 minutes decay" for a BWR and determined that under these conditions, the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits in 10 CFR Part 20.

We have considered the capabilities of the proposed gaseous radwaste treatment systems to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.

We have reviewed the applicant's quality assurance provisions for the gaseous radwaste systems, the quality group classifications used for system components, the seismic design applied to the design of the system, and of structures housing the radwaste systems. The design of the system and structures housing these systems meet the criteria as set forth in Regulatory Guide 1.143.

We have reviewed the provisions incorporated in the applicant's design to control releases due to hydrogen explosions in the gaseous radwaste system and conclude that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion or to withstand the effects of an explosion.

Based on the foregoing evaluation, we conclude that the proposed gaseous radwaste treatment system is acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the gaseous radwaste treatment system to the applicable regulations and Regulatory Guides referenced above.

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation."

2. 10 CFR §50.34a, "Design Objective for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Reactors."

3. 10 CFR §50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection."

6. 10 CFR Part 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

7. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)."

8. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs)."

9. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components in Light-Water-Cooled Nuclear Reactor Power Plants."

10. Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

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SECTION 11.4 SOLID WASTE MANAGEMENT SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

At the construction permit (CP) stage, ETSB reviews the design objectives, criteria, performance objectives, and description of the solid waste system (SWS) as given in the applicant's preliminary safety analysis report (PSAR). During the operating license (OL) stage of review, ETSB confirms the design accepted at the CP stage and evaluates the applicant's process control program and technical specifications in these areas.

1. a. The design objectives in terms of expected and design volumes of waste to be processed and handled, the wet and dry types of waste to be processed (e.g., sludges, resins, evaporator bottoms, and dry material such as contaminated tools, equipment, rags, paper, and clothing), the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the SWS are reviewed. The description of the SWS, the piping and instrumentation diagrams (P&IDs), and the process flow diagrams showing the methods of operation and factors that influence waste treatment are reviewed. The expected chemical content, flows and radionuclide concentrations of liquid wastes to be processed and handled by the SWS and the expected volumes to be returned to the liquid radwaste system for further treatment are reviewed.

b. The description of the methods for solidification (i.e., of removal of free water), the solidifying agent used, and a process control program to ensure a solid matrix are reviewed.

c. The description of the type and size of solid waste containers; the method of filling, handling, and monitoring for removable radioactive contamination; and provisions for decontamination, packaging and storage to meet applicable Federal regulations are 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.

2. The provisions for the onsite storage of solid wastes, the expected and design volumes, the expected radionuclide contents, and the design bases for these values are reviewed.

3. The quality group classifications of piping, and equipment and bases governing the classification chosen are reviewed.

4. Design provisions incorporated in the equipment and facility design to reduce leakage and facilitate operation and maintenance are reviewed.

5. Special design features, referenced topical reports, and previous experience with similar equipment and methods referenced in the SAR are reviewed.

6. The technical specifications proposed by the applicant for process and effluent control are reviewed at the operating license (OL) stage (FSAR).

SEB will provide an evaluation of the applicant's proposed seismic design of structures housing the solid radwaste systems.

II. ACCEPTANCE CRITERIA

1. ETSB will accept the solid waste treatment system design if the following conditions are met:

   a. The system design parameters are based on radionuclide concentrations and volumes consistent with reactor operating experience for similar designs and with the source terms of SRP Section 11.1.

   b. All wet wastes will be solidified prior to shipment offsite or there will be provisions to verify the absence of free liquid in the containers and to reprocess containers in which free liquid is detected in accordance with Branch Technical Position (BTP) ETSB 11-3 (Ref. 8).

   c. Solid waste containers, shipping casks, and methods of packaging meet applicable Federal regulations, e.g., 10 CFR Part 71, and wastes are to be shipped to a licensed burial site in accordance with applicable Commission and Department of Transportation regulations.

2. ETSB will accept the design capacity of the SWS if the following conditions are met:

   a. Processing equipment is sized to handle the design SWS inputs, e.g., the solid waste generation rates reviewed under I.1 of this SRP section without the need to ship bulk liquids or dewatered sludges or spent resins.

b. Onsite waste storage facilities provide sufficient storage capacity to allow time for short-lived radionuclides to decay prior to shipping in accordance with Branch Technical Position 11-3.

The bases for the storage time chosen should be given in the safety analysis report.

3. ETSB will accept SWS components and piping systems, and structures housing SWS components, designed in accordance with the provisions of Regulatory Guide 1.143 (Ref. 7).

4. ETSB will accept systems that contain provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Regulatory Guide 1.143 (Ref. 7) and Branch Technical Position (BTP), ETSB 11-3 (Ref. 8).

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this SRP section as may be appropriate for a particular case.

1. a. ETSB reviews the P&IDs and the process flow diagrams to determine system design, methods of operation, and parameters used in the design, i.e., expected and design flow rates, radioactivity concentrations, radionuclide distribution, and waste categories.

The system design and design criteria will be compared with the guidelines of Branch Technical Position (BTP) ETSB 11-3 and available data from operating plants of similar design.

b. ETSB compares the methods to be used to solidify liquids with experience gained from previous licensing reviews and with available data from operating plants employing similar methods. ETSB will review the process control programs to assure that the proposed solidification method is capable of solidifying the range of constituents expected to be present in the wastes. ETSB reviews the methods proposed to verify that all wet wastes have been immobilized or combined during solidification operations and will determine its acceptability considering (1) the ability of the technique to detect free, mobile, or uncombined liquids, (2) the procedure to be employed to solidify free liquids if detected, and (3) the effect of the method on operator exposures.

c. ETSB reviews the description of procedures for the packaging and shipment of solid wastes to an approved offsite burial facility, and verifies that the applicant makes definite commitments to following appropriate Federal and State regulations. ETSB compares the values given in the SAR for the

volumes and radionuclide content of solid wastes to be shipped offsite with data from operating plants of similar design and information from previous license applications.

2. ETSB compares the solid waste system design capacity with the design basis input waste volumes to determine whether the applicant has provided sufficient reserve capacity for greater-than-expected waste volumes which may occur as a result of anticipated operational occurrences. The inplant storage capacity is compared to the guidelines of (BTP) ETSB 11-3. The comparison will be based on the design criteria as stated in the SAR, on the availability of system components to handle surge flows, and on whether the storage facilities will provide onsite storage periods sufficient to permit the decay of short-lived radionuclides.

3. ETSB compares the quality group classifications and seismic design of the solid waste system and of the structures housing the system to the guidelines of Regulatory Guide 1.143. SEB reviews the seismic design criteria of structures housing the solid radwaste systems in accordance with the design guidance in Regulatory Guide 1.143. The consequences of failures of tanks containing radioactive liquids are evaluated under SRP Section 15.7.3.

4. ETSB compares equipment layout, design features, and mode of operation of the solid waste system to the guidelines of Regulatory Guide 1.143 and (BTP) ETSB 11-3.

5. At the OL stage ETSB reviews the technical specifications proposed by the applicant for process and effluent control. The reviewer will determine that the content and intent of the technical specifications prepared by applicant are in agreement with the requirement developed as a result of the staff's review. The review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

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 solid waste system (SWS) includes the equipment and instrumentation used for the solidification, packaging, and storage of radioactive wastes prior to shipment offsite for burial. The scope of the review of the SWS includes line diagrams of the system, piping and instrumentation diagrams (P&IDs), and descriptive information for the SWS and for those auxiliary supporting systems that are essential to the operation of the SWS. The applicant's proposed design criteria and design bases for the SWS, and the applicant's analysis of those criteria and bases have been reviewed. The capability of the proposed system to process the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with General Design Criterion 60, provisions

for the handling of wastes relative to the requirements of 10 CFR Parts 20 and 71 and of applicable DOT regulations, and the applicant's quality group classification and seismic design relative to Regulatory Guide 1.143, have also been reviewed. The applicant's proposed methods of assuring complete solidifcation have been reviewed. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the solid radwaste system to the regulations and the guides referenced above, as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed solid radwaste system is acceptable."

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation," and Appendix B, "Concentrations in Air and Water Above Natural Background."

2. 10 CFR §50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Reactors."

3. 10 CFR §50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection."

6. 10 CFR Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Materials Under Certain Conditions."

7. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components in Light-Water-Cooled Nuclear Reactor Power Plants."

8. Branch Technical Position ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," attached to SRP Section 11.4.

Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants

A. BACKGROUND

During normal operation of a nuclear power plant, radioactive materials are generated in the form of "wet" and "dry" wastes. Wet wastes, including spent bed resins, filter sludge, spent powdered resins, evaporator and reverse osmosis concentrates, and spent cartridge filter elements, normally result as byproducts from liquid processing systems. Dry wastes, including activated charcoal, HEPA filters, rags, paper, and clothing, normally result as byproducts from ventilation air and gaseous waste processing systems and maintenance and refueling operations. Wet and dry wastes will require processing in appropriate portions of the solid waste management system.

Compressible dry wastes such as contaminated rags, paper and clothing normally undergo a compaction process to reduce the volume of waste shipped offsite. Special provisions are needed to assure that contaminated airborne dusts are not released to the process area during compaction.

Wet wastes such as spent bed and powdered resins, filter sludge, and evaporator and reverse osmosis concentrates normally are solidified (i.e., combined with a suitable binder) prior to shipping, to render the waste immobile and thereby mitigate the consequences of potential ruptures to shipping containers.

Although there are a number of processes available which are capable of solidifying wet wastes under controlled conditions, there is a potential for free[1] liquids to remain in containers following solidification with the widely varying chemical species encountered during power plant operations. Based on the NRC staff's judgment, it is necessary that vendors and operators implement certain measures to (1) establish process parameters within which systems must be operated to obtain complete solidification, and (2) assure systems are operated within the established process parameters, or (3) have provisions to detect free liquid in containers prior to shipment offsite.

Following packaging, wastes are normally stored for decay of short-lived radionuclides and to accumulate sufficient wastes for a shipment offsite. Insofar as the continuous operation of the solid waste system is contingent upon storage space being available for the interim period between waste packaging and shipment offsite, consideration should be given to providing ample storage capacity to accommodate wastes during periods when shipments offsite are not possible, e.g., during labor strikes.

The criteria in Section B, below, provides adequate and acceptable design solutions for the concerns outlined above.

[1] For the purpose of this position paper, free water is defined as uncombined water not bound by the solid matrix.

This position paper sets forth minimum branch requirements and is not intended to prohibit the implementation of more rigorous design codes, standards, or quality assurance measures than those indicated.

B. BRANCH TECHNICAL POSITION

I. PROCESSING REQUIREMENTS

1. Dry Wastes

a. Compaction devices for compressible dry wastes (rags, paper, and clothing) should include a ventilated shroud around the waste container to control the release of airborne dusts generated during the compaction process.

b. Activated charcoal, HEPA filters, and other dry wastes which do not normally require solidification processing should be treated as radioactively contaminated solids and packaged for disposal in accordance with applicable Federal regulations.

2. Wet Wastes

a. Wet wastes such as spent bed and powdered resins, filter sludge, and evaporator and reverse osmosis concentrates should be rendered immobile by combining with a suitable binding agency (cement, urea formaldehyde, asphalt, etc.) to form a homogeneous solid matrix (absent of free water) prior to offsite shipment. Adsorbents such as vermiculite are not acceptable substitutes for binding agents.

b. Spent cartridge filter elements may be packaged in a shielded container with a suitable adsorber such as vermiculite, although it would be desirable to solidify the elements in a suitable binder.

II. ASSURANCE OF COMPLETE SOLIDIFICATION

Complete solidification of wet wastes should be assured by the implementation of a process control program or by methods to detect free liquids within container contents prior to shipment.

1. Process Control Program

a. Solidification (binding) agents and potential waste constituents should be tested and a set of process parameters (pH, ratio of waste to agent, etc.) established which provide boundary conditions within which reasonable assurance can be given that solidification will be complete.

b. The solid waste processing system (or liquid waste processing system, as appropriate) should include appropriate instrumentation and wet waste sampling capability necessary to successfully implement and/or verify the process control program described in a., above.

c. The plant operator should provide assurance that the process is run within the parameters established under a., above. Appropriate records should be maintained for individual batches showing conformance with the established parameters.

2. Free Liquid Detection

Each container filled with solidified wet wastes should be checked by suitable methods to verify the absence of free liquids. Visual inspection of the upper surface of the waste in the container is not alone sufficient to ensure that free water is not present in the container. Provisions to be used to verify the absence of free liquids should consider actual solidification procedures which may create a thin layer of solidification agent on top without affecting the lower portion of the container.

III. WASTE STORAGE

1. Tanks accumulating spent resins from reactor water purification systems should be capable of accommodating at least 60 days waste generation at normal generation rates. Tanks accumulating spent resins from other sources and tanks accumulating filter sludges should be capable of accommodating at least 30 days waste generation at normal generation rates.

2. Storage areas for solidified wastes should be capable of accommodating at least 30 days waste generation at normal generation rates. These storage areas should be located indoors.

3. Storage areas for dry wastes and packaged contaminated equipment should be capable of accommodating at least one full offsite waste shipment.

IV. ADDITIONAL DESIGN FEATURES

The following additional design features should be incorporated into the design of the solid waste system.

1. Evaporator concentrate piping and tanks should have heat tracing if the concentrates are likely to solidify at ambient temperatures.

2. Components and piping which contain radioactive slurries should have flushing connections.

3. Solidification agents should be stored in low radiation areas, generally less than 2.5 mr/hr, with provisions for sampling.

4. Tanks or equipment which use compressed gases for transport or drying of resins or filter sludges should be vented directly to the plant ventilation exhaust system which includes HEPA filters as a minimum. The vent design should prevent liquids and solids from entering the plant ventilation system.

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SECTION 11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

ETSB reviews the following information in the applicant's safety analysis report (SAR):

1. The design objectives and design criteria for the process and effluent radiological monitoring and sampling system are reviewed. The review includes the identification of the process and effluent streams to be monitored or sampled, the purpose of each monitoring or sampling function provided, and the parameters to be determined through monitoring or sampling (e.g., gross beta-gamma concentrations, radionuclide distribution, or quantities of specific radioisotopes).

2. The system description for the process and effluent radiological monitoring and sampling system is reviewed. The review includes (a)* description of instrumentation provided, including redundancy, range, calibration, sensitivity, alarm/trip setpoints, independence, and diversity of components for normal operations, anticipated operational occurrences, and postulated accidents; (b)* location of monitors and direct readouts; (c)* location of sampling points and sampling stations; (d) calculation of radioactivity concentrations to be monitored or sampled for normal operations, anticipated operational occurrences, and postulated accidents; (e) measurements or determinations to be made (e.g., gross beta-gamma concentration or measurement of specific radionuclides); (f)* types and locations of annunciations and alarms and the actions initiated by each, (g) provisions for purging sample lines, input volumes to waste collection systems, and sampling frequency, (h) expected relationships between monitoring and sampling results and plant operations; (i)* descriptions or procedures for calibration, maintenance, and inspection of monitoring instrumentation; (j) layout drawings, piping and instrumentation diagrams (P&IDs), and process flow diagrams.

3. The technical specifications proposed by the applicant for process and effluent control are reviewed at the operating license (OL) stage (FSAR).

* Final Safety Analysis Report (FSAR) only.

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.

ICSB will review the seismic design, redundancy, and emergency power supply for process and effluent radiological monitors required to actuate safety features which prevent or mitigate the consequences of accidents which could result in offsite exposures comparable to the guidelines of 10 CFR Part 100.

AAB will review specification and criteria for monitoring instrumentation and postulated accidents.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. Provisions should be made to monitor or sample all normal and potential pathways for release of radioactive materials to the environment in conformance with General Design Criterion 64. Guidance is provided in Regulatory Guides 1.21, 1.97 and 4.15.

   a. The gaseous and liquid process streams or effluent release points should be monitored and sampled according to Tables 1A and 1B.

TABLE 1A

Provisions for Monitoring and Sampling Gaseous Streams

No.	Process Systems	Reactor Type	Monitor Provisions In Process Cont[I]	Monitor Provisions In Effluent ACF[J]	Monitor Provisions In Effluent Cont[I]	Sample Provisions In Process Grab[K]	Sample Provisions In Effluent Grab[K]	Sample Provisions In Effluent Cont[I]
1.	Waste Gas Holdup System[a]	P&B	N	N	(N)		(NT)	(I)
2.	Condenser Evacuation System[b]	P&B	N	(N)	(N)	I	(NT)	(I)
3.	Vent&Stack Release Pt. System[c]	P&B			N		NIT	I
4.	Containment Purge Systems[d]	P&B	N	N[X]	(N)	I	(NT)	(I)
5.	Aux. Bldg. Ventilation System	P&B			(N)	I	(NT)	(I)
6.	Fuel Storage Area Vent. Syst.[e]	P&B	(N)	(N)[X]	(N)	I	(NT)	(I)
7.	Radwaste Area Vent. Systems	P&B			(N)	I	(NT)	(I)
8.	Turb. Gland Seal Cond. Vent System	P&B			(N)	I	(NT)	(I)
9.	Mech. Vacuum Pump Exhaust (Hogging) System	P&B			(N)	I	(NT)	(I)
10.	Evaporator Vent Systems	P&B			(N)	I	(NT)	(I)
11.	Pre-treatment Liquid Radwaste Tank Vent Gas Systems	P&B			(N)	(I)	(NT)	(I)
12.	Flash Tank and Steam Generator Blowdown Vent Systems	P			(N)	I	(NT)	(I)
13.	Turbine Bldg. Vent. Systems	B			(N)	I	(NT)	(I)
14.	Pressurizer & Boron Recovery Vent Systems	P			(N)	I	(NT)	(I)

TABLE 1B

Provisions for Monitoring and Sampling Liquid Streams

No.	Process Systems	Reactor Type	Monitor Provisions			Sample Provisions		
			In Process	In Effluent		In Process	In Effluent	
			Cont[i]	ACF[j]	Cont[i]	Grab[k]	Grab[k]	Cont[i]
1.	Liquid Radwaste (Batch) Effluent System	P&B	G	G	G	GR	GRT	
2.	Liquid Radwaste (Cont.) Effluent System	P&B	G	G	G		GRT	GR
3.	Service Water System	P&B			G		GRT	G
4.	Component Cooling Water System[f]	P&B	G	(G)[x]	(G)	GR	GRT	(GR)
5.	Spent Fuel Pool Treat. Syst.[g]	P&B	G	(G)	(G)	GR	(GRT)	(GR)
6.	Equip. & Floor Drain Collection and Treatment Systems	P&B		(G)	(G)		(GRT)	(GR)
7.	Phase Separator Decant & Holding Basin Systems	P&B		(G)	(G)		(GRT)	(GR)
8.	Chemical & Regeneration Solution Waste Systems	P&B		(G)	(G)		(GRT)	(GR)
9.	Laboratory & Sample System Waste Systems	P&B		(G)	(G)	GR	(GRT)	(GR)
10.	Laundry & Decontamination Waste Systems	P&B		(G)	(G)		(GRT)	(GR)
11.	Resin Slurry, Solidification & Baling Drain Systems	P&B	G		(G)		(GRT)	(GR)
12.	Radwaste Liquid Tanks (outside the buildings)	P&B			(G)	GR	(GRT)	(GR)
13.	Storm & Underdrain Water Syst.	P&B					(GRT)	
14.	Tanks and Sumps inside Contmt.	P&B		(G)	(G)		(GRT)	(GR)
15.	Boron Recovery System Liquid Effluent	P		(G)	(G)		(GRT)	(GR)
16.	Steam Generator Blowdown (Batch) Liquid Effluent System	P	G	G	G	GR	(GRT)	(GR)
17.	Steam Generator Blowdown (Cont) Liquid Effluent System	P	G	G	G		(GRT)	(GR)
18.	Sec. Coolant Treatment Waste & Turbine Bldg. Drn. Systems	P		(G)	(G)		(GRT)	(GR)
19.	Ultrasonic Resin Cleanup Waste Systems	B		(G)	(G)		(GRT)	(GR)
20.	Non-Contaminated Waste Water & PWR Turb. Bldg. Cold Drn. Syst.	P&B					(GRT)	(GR)

Notes for Table 1A and Table 1B

a - for example, offgas storage tank system, cover gas decay systems, chilled charcoal absorption systems, offgas cryogenic units and delay pipes.

b - for example, main condenser steam jet air ejector systems and vacuum pump systems.

c - for example, free standing stacks, roof vents, building vents, exhausters, process vents, ventilation vents.

d - for example, containment relief systems, containment normal purge, containment low purge, containment leak sample systems, drywell purge, cleanup purges.

e - includes spent fuel pool and refueling pool ventilation systems, if separate from the fuel storage area ventilation system.

f - also called closed cooling water systems and component cooling loop systems.

g - includes refueling pool cleanup systems.

h - includes suppression tanks, reactor drain tanks, equipment and drain sumps collecting leakage, drainage, sampling and condensate.

i - continuous radiation monitor.

j - automatic control feature. The continuous effluent radiation monitor (see note x) should be equipped to alarm at a set point established in the technical specifications and automatically terminate effluents in the discharge line by closing an isolation valve (see II.3.d).

k - sample point available to obtain grab samples for laboratory analyses as indicated by notations.

l - continuous sampler (see II.1.b).

x - the automatic control feature may be provided by the process continuous radiation monitor.

Notations for Table 1A and Table 1B

P - Typical system names applicable to pressurized water reactors.

B - Typical system names applicable to boiling water reactors.

N - Noble gas radioactivity.

I - Radioiodine radioactivities and radioactivity of materials in particulate form and alpha emitters.

T - Tritium radioactivity (carbon-14 analysis in gaseous effluents may be considered).

G - Gross radioactivity.

R - Principal identification and concentration of radionuclides and alpha emitters.

( ) These provisions are required only for systems not monitored, sampled or analyzed (as indicated) prior to release by downstream provisions.

b. For both BWRs and PWRs, liquid wastes and confined volumes of gaseous waste should be sampled batchwise prior to release, in accordance with Regulatory Guide 1.21. Continuous gaseous effluent monitors for open structures, such as a PWR turbine building or treated liquid waste tanks outside the buildings are not required. For liquid and gaseous effluents that cannot be practicably monitored or sampled batchwise, one of the following alternate methods of representative sampling should be provided:

(1) A continuous proportioning (liquid effluent) sampling system with at least two sample collection tanks. The system should be designed to provide a fixed or measured flow ratio of the sample collected to the sampled stream discharge, or

(2) A periodic automatic grab (liquid effluent) sampling system with at least two sample collection tanks. The system should be designed to collect a fixed volume of sample at a rate proportional to the measured flow in the sampled stream discharge, or

(3) A continuous composite (radioactive materials in gaseous effluents other than noble gases) sampling system with replaceable high efficiency particulate filter and radioiodine absorber. The system should be designed to automatically composite samples at a fixed or measured flow ratio of the sample throughput to the sampled stream discharge, and

a periodic analyses frequency for the collected sample should be specified in the technical specifications.

2. Provisions should be made to monitor or sample radioactive waste process systems in conformance with General Design Criterion 63.

a. Provisions should be made to assure representative samples from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks (collection or sample test tanks) should be capable of recirculating two tank volumes in approximately eight hours. For process stream samples, provisions should be made for purging sample lines and for reducing plateout in sample lines. Provisions for sampling from ducts and stacks should be in agreement with ANSI N13.1.

b. Provisions should be made to collect samples from process waste streams at central sample stations to reduce leakage, spillage, and radiation exposures to operating personnel in accordance with SRP Section 9.3.2.

c. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system.

3. Instrumentation, sampling and monitoring provisions should conform to the following:

   a. Sampling frequencies, required analyses, instrument alarm/trip set points, calibration and sensitivities, and provisions to composite samples for low-level analyses should be in conformance with Regulatory Guide 1.21. Sampling frequencies and required analyses should be given in the plant technical specifications; these provisions will be reviewed at the OL stage.
   b. Provisions for the necessary instrumentation and facilities to perform gross beta-gamma and gross alpha measurements, isotopic analyses, and other routine analyses should be in conformance with Regulatory Guide 1.21.
   c. Provisions should be made to perform routine instrument calibration, maintenance, and inspections. The frequencies of such actions should be given in the plant technical specifications. The provisions will be reviewed at the OL stage. Provisions should also be made to replace or decontaminate monitors without opening the process system or losing the capability to isolate the effluent stream.
   d. Isolation valves with automatic control features should fail in the closed position. The release rates should be established in the plant technical specifications. Instrumentation provisions for automatic termination of releases should conform to the design guidance contained in Appendix A to this chapter.

4. The process and effluent radiological monitoring and sampling system provisions for postulated accidents will be acceptable if normal gaseous effluent paths are provided with supplemental monitoring equipment capable of monitoring postulated accident releases in accordance with ANSI Draft Standard N13/42 WG6 and Regulatory Guide 1.97.

5. The PERMSS provisions for monitoring liquid effluents for postulated accidents will be acceptable if both normal and postulated accident liquid effluents are discharged in the batch mode, if administrative procedures are in effect to minimize inadvertent or accidental releases of radioactive fluids, and if the normal liquid effluent monitors provide automatic termination of releases in the event that established effluent control levels are exceeded. Only normal operation effluent monitors are needed to meet these acceptance criteria.

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 review of the process and effluent radiological monitoring and sampling system, ETSB will compare the listing of process and effluent monitors contained in the SAR with the principal release points identified in SRP Section 11.2 and

SRP Section 11.3 to assure that all major process streams and release pathways are being monitored during normal operation and anticipated operational occurrences. The review includes the following:

a. The location of probes, sample stations, and the bases for the selection of these sample points are compared with the general principles for obtaining valid samples of airborne radioactive materials, the methods and materials for gas and particulate sampling, and guides for sampling from ducts and stacks contained in ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities" (Ref. 2).

b. The equipment, piping, and description of sampling methods to assure representative sampling will be compared with the guidelines given in Regulatory Guide 1.21.

c. ETSB will independently calculate on an audit basis the radiation levels and concentrations in the process and effluent stream using the models of NUREG-0016 or NUREG-0017 to verify the expected levels (Refs. 5 and 6).

d. ETSB will compare the sampling frequencies, types of analyses required, and monitoring instrument sensitivities with those recommended in Regulatory Guide 1.21. At the OL stage, ETSB will compare the applicant's monitoring instrumentation specifications and performance criteria with those contained in ANSI N13.10-1974 (Ref. 7) and in Appendix A to this chapter.

e. In the review of the P&IDs for the waste treatment system, ETSB will verify that release points for radioactive material have provisions for automatic termination of releases in the event they exceed a predetermined level. Instrumentation provisions for automatic termination of releases will be compared with the design guidance contained in Appendix A to this chapter.

f. ETSB will review the location of the monitors shown on the P&IDs and the location of readouts, annunciators, and alarms discussed in SAR Chapter 7 to assure that the operator will be advised of system performance and effluent releases consistent with the release limits specified in the plant technical specifications.

g. ETSB will compare the calibration methods and frequency with the guidelines in Regulatory Guide 1.21 (FSAR).

h. ETSB will assure that provisions are included in the design for replacing detectors or decontaminating the monitors without opening the process system or losing the capability to isolate the system or divert the effluent to a standby treatment system.

i. In the review of the radiation levels and concentrations of radioactive material expected from postulated accidents, ETSB will compare the applicant's values with values contained in ANSI N13/42 WG6 (Ref. 3).

j. In the review of the design and operation of the monitoring systems for postulated accidents, ETSB will compare the applicant's design with the guidance contained in ANSI N13/42 WG6.

k. ETSB will review special features, applicable topical reports, and data referenced in the SAR on a case-by-case basis.

2. ETSB reviews the technical specifications proposed by the applicant for process and effluent radiological monitoring and sampling at the OL stage. The reviewer determines that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements developed as a result of the staff's review. The review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

IV. EVALUATION FINDINGS

ETSB 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 process and effluent radiological monitoring and sampling systems include the instrumentation for monitoring and sampling contaminated liquid, gaseous, and solid waste process and effluent streams. Our review included the provisions proposed to sample and monitor all station effluents in accordance with General Design Criterion 64, the provisions proposed to provide automatic termination of effluent releases and assure control over discharges in accordance with General Design Criterion 60, the provisions proposed for sampling and monitoring plant waste process streams for process control in accordance with General Design Criterion 63, the provisions for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guide 1.21, and the provisions for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems, and ventilation systems, and the location of monitoring points relative to effluent release points on the site plot diagram.

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the process and effluent radiological monitoring and sampling systems to the applicable regulations and guides, as indicated above, as well as to staff technical positions and industry standards. Based on our evaluation, we find the proposed systems to be acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Material to the Environment"; Criterion 63, "Monitoring Fuel and Waste Storage"; and Criterion 64, "Monitoring Radioactivity Releases."

2. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," American National Standards Institute (1969).

3. ANSI N13/42 WG6, Draft, "Performance Specifications for Reactor Emergency Monitoring Instrumentation," American National Standards Institute (1974).

4. Regulatory Guide 1.21, "Measuring and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

5. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," (PWR-GALE Code).

6. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," (BWR-GALE Code).

7. ANSI N13.10-1974, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents" (1974).

8. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant Conditions During and Following an Accident."

9. Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment."

10. Standard Review Plan Appendix 11.5-A, "Design Guidance for Radiological Effluent Monitors."

APPENDIX 11.5-A DESIGN GUIDANCE FOR RADIOLOGICAL EFFLUENT MONITORS

A. Background

The primary design function of a radiological effluent monitor is the detection and measurement of radioactive materials released in gaseous or liquid effluent streams of light-water-cooled nuclear power reactors. An additional design function of some monitors is that of providing a signal to automatically terminate or otherwise modify the effluent stream. Examples are the termination of exhaust air flow by closure of containment ventilation or purge isolation valves and diversion of building ventilation exhaust streams from an untreated discharge path to an alternative treatment system, such as a standby gas treatment system for a BWR plant.

Depending on plant design and onsite meteorology, such an action may be necessary to mitigate the consequences of a design basis accident (DBA). The need for such mitigation is determined by calculating offsite doses that would result from the DBA. In other plant designs, radiological effluent monitors are used to actuate systems to modify or terminate releases for other purposes, e.g., to terminate releases due to anticipated operational occurrences to assure that offsite doses are maintained within the limits specified in the plant Technical Specifications.

The design and quality assurance criteria applied to the design, procurement, installation, testing, and operation of radiological effluent monitors installed in light-water-cooled nuclear power reactors should provide assurance that the monitors will perform all of their design functions.

If the DBA analysis shows that the actuation of an engineered safety feature (ESF) system is required to mitigate the consequences of a DBA, and that a signal from a radiological effluent monitor is necessary to actuate the ESF system, then the monitor should be designed and qualified to the design and quality assurance criteria applicable to the ESF system. Conversely, if an automatically functioning device or system is used to reduce radioactive releases to assure maintaining offsite doses within the limits of the plant Technical Specifications (i.e., not for the purpose of mitigating the consequences of a DBA), then a monitor providing the actuation signal should be designed and qualified to criteria consistent with those of the actuated system.

This appendix does not establish, nor does it change in any manner, the design and quality assurance criteria established elsewhere for ESF or ESF-related systems or monitors.

The design guidance set forth in this Appendix provides reasonable assurance that monitors used to provide initiation signals for actuation of systems to control the release of radioactive materials in effluents. but not required to mitigate the consequences of a DBA, are designed, constructed, installed, tested, and maintained on a level commensurate with their intended function.

This Appendix sets forth minimum requirements and is not intended to prohibit the implementation of equivalent design codes, standards, or quality assurance measures other than those indicated herein.

B. Definitions

Radiological Effluent Monitor: A device which removes a representative sample from the effluent stream, detects and quantitatively measures the radioactive materials present in the sample, discharges the sampled medium back to the effluent stream, and transmits the measurement data to a central point.

Monitoring System: A system consisting of one or more remote monitors, a centrally located cabinet or console where data from the monitors is received, recorded, and displayed, and the necessary interconnecting cables, power supplies, pumps, motors, alarms, recorders, display panels, relays, and other auxiliary components.

C. Design Guidance

Design and quality assurance criteria for radiological effluent monitors should be consistent with the design and quality assurance criteria applicable to the systems which are actuated by a signal from the monitors.

Monitors providing signals for the actuation of engineered safety feature (ESF) systems should be designed and qualified to the design and quality assurance criteria applicable to ESF systems. Criteria for ESF-related monitors are found in the appropriate Standard Review Plans under Section 7. This position does not affect or modify existing criteria for ESF-related systems.

Monitors providing signals for the actuation of non-ESF systems should be designed and qualified to the design and quality assurance criteria applicable to the actuated system or to the criteria shown in Table I, below.

D. Implementation

The purpose of this section is to provide information to applicants and licensees regarding the staff's plans for utilizing this Appendix.

Except in those cases in which the applicant proposes an alternate method for complying with specified portions of the Commission's regulations, the criteria described herein will be used in the evaluation of applications docketed after July 1, 1979, as follows:

1) Preliminary Design Approval (PDA) application reviews and Preliminary Duplicate Design Approval (PDDA) application reviews.

2) Final Design Approval, Type 2, (FDA-2) application reviews and Final Duplicate Design Approval, Type 2, (FDDA-2) application reviews. Final Design Approval, Type 1, (FDA-1) or Final Duplicate Design Approval, Type 1 (FDDA-1) applications docketed after July 1, 1979, should conform to the criteria of the corresponding PDA or PDDA applications, respectively.

3) Manufacturing License (ML) application reviews.

4) Construction Permit (CP) application reviews, except that portions of CP applications docketed after July 1, 1979, which reference standard designs (i.e., PDA's, FDA-1's, FDA-2's, PDDA's, FDDA'1's, FDDA-2's, or ML's) and replicate plant applications docketed after July 1, 1979, should conform to the criteria of the referenced design.

5) Operating License (OL) applications should conform to the criteria of the corresponding CP applications.

These criteria do not apply to operating plants.

If an applicant wishes to use this guidance in developing submittals for applications docketed on or before July 1, 1979, the pertinent portions of the application will be evaluated on the basis of this Standard Review Plan.

TABLE I

DESIGN GUIDANCE FOR RADIOLOGICAL EFFLUENT MONITORS
(INSTRUMENTATION INSTALLED IN LIGHT WATER COOLED NUCLEAR POWER PLANTS)

Category	Design Criteria	Quality Assurance Criteria
Effluent Radiological Monitoring Instrumentation providing a signal for the actuation of a system used to reduce releases of radioactive materials in effluents within limits specified in plant Technical Specifications. (Not required to initiate actuation for an ESF system).	Review: Reviewed under Standard Review Plan 11.5 Reviewed by: Effluent Treatment Systems Branch (ETSB) Criteria: Manufacturer's Standard per ANSI N13.10-1974.	Review: Reviewed under Standard Review Plan 11.5 Reviewed by: Effluent Treatment Systems Branch (ETSB) Criteria: Quality Assurance set forth in Regulatory Guide 1.143, Section IV.

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

SECTION 12.1 ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) related to assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA) are reviewed:

1. POLICY CONSIDERATIONS
   a. Management policy with respect to designing and constructing the plant (preliminary safety analysis report, PSAR) and with respect to operating the plant (final safety analysis report, FSAR) and the planned organizational structure (PSAR).
   b. The applicable activities carried on by the individuals in management having responsibility for radiation protection (PSAR).
   c. Information describing the implementation of policy, organizational, training, and design review guidance or requirements provided in Regulatory Guide 8.8, Regulatory Guide 8.10, Regulatory Guide 1.8, and in 10 CFR Part 20. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

2. DESIGN CONSIDERATIONS
   a. Facility and equipment design considerations (PSAR) (but not design features, as in SAR Section 12.3).
   b. The description of the use of dose assessment (SAR Section 12.4) and the resultant man-rem doses associated with various work functions for evaluating the facility design relative to assuring that ORE will be ALARA (PSAR).
   c. The description of how experience from past designs, from various design guidelines, such as Regulatory Guide 8.8, and from operating plants is used to develop improved radiation protection designs (PSAR).
   d. Information describing the implementation of the design guidelines of Regulatory Guide 8.8 and other industry developed design guidance that includes the ALARA criteria. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

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.

3. OPERATIONAL CONSIDERATIONS

a. The techniques and methods that will be used for developing detailed operational plans and procedures (PSAR).

b. The description of how procedures for operation may impact on and interact with the plant design and on the design process (PSAR).

c. The use of operating plant experience in planning the operational considerations for new plant designs (PSAR).

d. The criteria and conditions under which various exposure reduction procedures and techniques are implemented (FSAR).

e. Information describing the implementation of radiation protection programs, and operational guidance of Regulatory Guide 8.8 and Regulatory Guide 8.10. Information describing alternatives, if such as proposed (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.1 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. Specific acceptance criteria for these areas of review are as follows:

1. POLICY CONSIDERATIONS

Acceptability will be based on evidence in the PSAR that a policy for assuring that ORE will be ALARA has been formulated, described, displayed, and will be implemented in accordance with Regulatory Guide 8.8, Regulatory Guide 8.10, C.1, and Regulatory Guide 1.8. This includes designating and assigning a specific individual responsibility and authority for implementing ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the Regulatory Guides and industry standards.

2. DESIGN CONSIDERATIONS

Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with Regulatory Guide 8.8, including incorporation of: measures for reducing the need for maintenance; measures to improve the accessibility to components requiring periodic maintenance or inservice inspection; measures to reduce the production, distribution and retention of activated corrosion products throughout the primary system; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; feedback of experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a comparison with the Regulatory Guide.

3. OPERATIONAL CONSIDERATIONS

Acceptability will be based on evidence that the applicant has a program to develop plans and procedures in accordance with Regulatory Guide 8.8, which can incorporate the experiences obtained in facility operation into facility and equipment design and into operations planning and which will implement specific exposure control techniques. In the FSAR, acceptance of criteria and conditions under which various exposure

reduction procedures and techniques are implemented will also be based on Regulatory Guides 8.8 and 8.10, C.2. Alternative proposed plans and procedures will be evaluated on the basis of a comparison with the Regulatory Guides.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2. RAB reviews the management policy and the planned organizational structure to determine how the guidance given in Regulatory Guides 8.8, 8.10 and 1.8 will be implemented or whether alternatives have been proposed. The organization structure review includes determining if the individuals responsible for the radiation protection program are on a level in management that will assure implementation of management's commitment for assuring that ORE will be ALARA. Any problems concerning the review of organizational structure, as related to the radiation protection manager will be communicated to Quality Assurance Branch, which has primary review responsibility for this item, in Chapter 13. The reviewer uses ANSI N18.7-1972 for additional guidance on acceptable operating organizations.

The reviewer evaluates information in this section to determine if the organizational structure provides a mechanism for the radiation protection manager and his organization interacting with design groups to assure that methods and techniques for reducing ORE will be incorporated in the design of the plant. Where the future plant radiation protection manager has not been selected, design review should be accomplished in accordance with the guidance of Regulatory Guide 8.8, or a proposed alternative. The reviewer determines from information furnished by the applicant that he has incorporated previously accepted design features and has used operating experience to improve on the design of the plant with regard to assuring that ORE will be ALARA. The reviewer compares the material in this section with the requirements of 10 CFR Part 20 and the guidelines of Regulatory Guides 8.8 and 8.10.

Based on the review RAB may request additional information or request the applicant to modify his submission in order to meet the acceptance criteria given in Subsection II.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's Safety Evaluation Report. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

The applicant provides a management commitment to assure that (plant name) will be designed, constructed, and operated in a manner consistent with Regulatory Guides 8.8, 8.10, and 1.8. The (title of person or group, e.g., Plant Health Physicist and staff) periodically reviews, updates, and modifies all plant design features, as well as all operating and maintenance features, using exposure data and experience gained from operating nuclear power plants.

This is done to assure that occupational exposures will be kept as low as is reasonably achievable in accordance with Regulatory Guide 8.8 criteria.

The objective of the plant radiation protection design is to maintain individual doses and total man-rem doses to plant workers, including construction workers, and to members of the general public as low as is reasonably achievable, and to maintain individual doses within the limits of 10 CFR Part 20. Within restricted areas all plant sources of direct radiation and airborne radioactive contamination are considered in our review. In unrestricted areas within the site boundary, this chapter considers radiation emitted from sources contained in the plant facilities and effluents from those facilities.

(Utility) will incorporate the following facility and equipment design considerations at (plant name) in order to satisfy the above listed radiation protection design objectives. (List several design considerations used.) These design considerations conform with the guidelines of Regulatory Guide 8.8 and are acceptable.

Operating and maintenance personnel follow specific plans and procedures in order to assure that "as low as is reasonably achievable" goals are achieved in the operation of the plant. High radiation exposure operations are carefully preplanned and carried out by personnel well-trained in radiation protection and using proper equipment. During such maintenance activities, personnel are monitored for exposure to radiation and contamination. Upon completion of major maintenance jobs, personnel radiation exposures are evaluated in a timely manner and compared with predicted man-rem exposures. The results are used to make changes in future job procedures and techniques. The management staff annually reviews radiation exposure trends to determine major changes in problem areas, and which worker groups are accumulating the highest exposures. The staff uses these reports to recommend design modifications or changes in plant procedures. These practices conform with those in Regulatory Guides 8.8 and 8.10 and are acceptable.

V. REFERENCES

1. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

2. 10 CFR Part 20, "Standards for Protection Against Radiation."

3. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Practicable."

4. Regulatory Guide 1.8, "Personnel Selection and Training."

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

6. ANSI N18.7, "Administrative Control for Nuclear Power Plants," American National Standards Institute.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
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SECTION 12.2 RADIATION SOURCES

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) relating to radiation sources are reviewed as related to in-plant radiation protection:

1. CONTAINED SOURCES

The description of the sources of radiation that are the basis for the radiation protection program, as used in the shield design calculations (SAR Chapter 11 contains the description for sources contained in equipment of the radioactive waste management systems). This should include isotopic composition, location in the plant, source strength and source geometry, and the basis for the values (preliminary safety analysis report (PSAR), and update in the final safety analysis report (FSAR). The descriptions should include any required byproduct, source, and special nuclear materials.

2. AIRBORNE RADIOACTIVE MATERIAL SOURCES

The description of the sources of airborne radioactive material considered in the design of the ventilation systems, as used for design of personnel protective measures and for dose assessment. (SAR Chapter 11 contains the description for airborne sources that have to be considered for their contribution to the plant effluent releases, through equipment of the radioactive waste management systems or the plant ventilation system.) The descriptions should include a tabulation of the calculated concentration of radioactive material by nuclides expected during normal operation and anticipated operational occurrences for equipment cubicles, corridors, and operating areas normally occupied by operating personnel and should include models and parameters for the calculations (PSAR and update in FSAR).

In the event of new and unused designs, ETSB will, on request, assist RAB in reviewing the sources and source term design bases.

II. ACCEPTANCE CRITERIA

Specific acceptance criteria for these areas of review are as follows:

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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.

All sources of radiation that necessitate designed shielding, special ventilation designs, special storage locations and conditions, traffic or access control considerations, special plans and procedures, monitoring equipment, etc., shall be described to the degree needed for the shielding codes used in the design process, for establishing related facility design features, for plans and procedures development, for assessment of occupational radiation exposures, and for equipment specifications. For contained sources, the description should include plan scale drawings of each floor of the plant on which all the sources of significance are shown and identified in a manner that can be easily related to tables containing the pertinent and necessary quantitative source parameters. Their position should be accurately indicated, as well as the approximate size and shape. Airborne sources that are created by leakage, by opening formerly closed containers, by storage of leaking fuel elements, etc., shall be identified by location and magnitude, in a manner useful for designing appropriate ventilation systems and in specifying appropriate monitoring systems. The assumptions made in arriving at quantitative values for these various sources should be specified, either in this section or by reference to Chapter 11 of the SAR.

Shielding and ventilation design fission product source terms will be acceptable if developed using these bases:

(1) an offgas rate of 100,000 μCi/sec after 30 minutes delay for BWRs.

(2) 0.25% fuel cladding defects for PWRs.

Coolant and corrosion activation products source terms should be based on applicable reactor operating experience. The buildup of activated corrosion products in various components and systems should be quantified. Any allowances made in design source terms for such buildups should be explained. Neutron and prompt gamma source terms should be based on reactor core physics calculations and applicable reactor operating experience.

The tables of quantitative source parameters, which can be placed in Chapter 12 or referenced to Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation or in a ventilation system design. In addition, the quantities will be acceptable if the reviewer can confirm specific values given in the tables from the assumptions given along with common tables of nuclide decay schemes, gamma ray energies, etc. ANSI N237 and Regulatory Guide 1.112 will be used for acceptance of coolant and corrosion activation products source terms. These documents provide guidance on estimating effluent source terms. The fuel cladding defect (offgas rate for BWRs), which forms the basis of the calculation of effluent fission product source terms, is a lower value in these documents, than the 0.25% (100,000 μCi/sec for BWRs) which should be used as the shielding source terms as presented above.

For byproduct, source, and special nuclear materials, 10 CFR Parts 30, 40 and 70 can be used as guidance.

III. REVIEW PROCEDURES

The reviewer determines whether the source term design bases are consistent with the acceptance criteria. Then, the reviewer determines whether source strengths, concentrations of airborne radioactivity, and quantitative source descriptions are consistent with the assumptions made and the methods used by the applicant. Locations of the contained sources relative to shield walls, occupied areas, traffic pathways, in-service inspection points, sampling stations, controls, etc., are examined for any potential problems in assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). Based on the review, RAB may request additional information or request the applicant to reevaluate his analysis for the purpose of modifying those areas which do not meet the acceptance criteria given in Subsection II.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's Safety Evaluation Report. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings.

This section contains a description of the sources of contained and airborne radioactivity used as inputs for the dose assessment and for the shielding and ventilation designs. Also included are the assumptions made by the applicant in arriving at quantitative values for these contained and airborne source terms.

Inside the containment during power operation, the greatest potential for personnel dose during operation is due to nitrogen-16, noble gases, and neutrons. Outside the containment and after shutdown inside the containment the primary sources of personnel exposure are fission products from fuel clad defects, and activation products, including activated corrosion products. Almost all of the airborne radioactivity within the plant is due to equipment leakage. The fission product source terms are based on 0.25% fuel cladding defects (PWRs). The coolant and corrosion activation product source terms are based on operating experience from reactors of similar design; allowances are included for the buildup of activated corrosion products. Neutron and prompt gamma source terms are based on reactor core physics calculations and operating experience from reactors of similar design. Other parameters used, as well as a complete description of source term development, are contained in Chapter 11. The source terms presented are comparable to estimates by other applicants with similar designs and are acceptable.

The applicant has provided a tabulation of the maximum expected radioactive airborne concentrations in equipment cubicles, corridors, and operating areas, due to equipment leakage. The bases for these leakage calculations are in accordance with Ref. 5.

The ventilation system will be designed to provide (number) volume changes per hour in occupied areas which may contain significant airborne activity. Air will be routed from areas of low potential airborne contamination to areas of increasing potential airborne contamination. The resulting estimated airborne radioactivity concentrations in frequently

occupied areas will be a small fraction of 10 CFR Part 20 limits and are acceptable. The source terms used to develop these airborne concentration values are comparable to estimates by other applicants with similar designs and are acceptable.

V. REFERENCES

1. 10 CFR Part 70, "Special Nuclear Material."

2. 10 CFR Part 40, "Source Material."

3. 10 CFR Part 30, "Byproduct Material."

4. ANSI N237, Final Draft by Working Group 18.1, "Source Term Specification," American National Standards Institute.

5. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors."

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SECTION 12.3 RADIATION PROTECTION DESIGN FEATURES

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) relating to radiation protection design features are reviewed:

1. FACILITY DESIGN FEATURES
   a. In the preliminary safety analysis report (PSAR), the description of equipment and facility design features used for assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA).
   b. The radiation zone designations, including zone boundaries for both normal operational and refueling conditions (PSAR and update in the final safety analysis report, FSAR).
   c. The illustrative examples of facility design features of the equipment, components, and systems listed in Sections 12.1.3 and 12.3.1 of "Standard Format and Content..." (Ref. 1) including the scaled layout and arrangement drawings of the facility showing all source locations and the other design details requested in Section 12.3.1 of the "Standard Format..." (PSAR and update in FSAR). Shield wall thicknesses for all shielded spaces should be specified on the drawings or provided in separate tables.
   d. The description of facilities and equipment for handling and use of sealed and unsealed special nuclear, source, and byproduct materials (PSAR and update in FSAR).
   e. Information describing implementation of Regulatory Guide 8.8 guidelines on facility and equipment design and layout. Information describing alternatives if such are proposed (PSAR and update in FSAR).

2. SHIELDING
   a. The shielding to be provided for each of the radiation sources identified in SAR Chapter 11 and Section 12.2, including the design criteria for penetrations and the shield material used (PSAR and update in FSAR). (Note item I.1.c above)
   b. The description of the methods by which the shield parameters were determined, including pertinent codes, assumptions, and techniques used or to be used in the calculations (PSAR and update in FSAR).

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 description of any special protective features that use shielding, geometric arrangement, or remote handling to assure that ORE will be ALARA (PSAR and update in FSAR).

d. Information describing implementation of Regulatory Guides 1.69 and 8.8 (regarding special protective features). Information describing alternatives, if such are proposed (PSAR and update in FSAR).

3. VENTILATION

a. The description of the personnel protection features incorporated in the ventilation system designs called for in Section 12.3.3 of "Standard Format and Content..." (PSAR and update in FSAR).

b. Illustrative example of the air cleaning system design (PSAR and update in FSAR).

c. Information describing any application of Regulatory Guide 1.52 (particularly Section C.4 & 5) and Regulatory Guide 8.8. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

4. AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

a. The description of the fixed area radiation and continuous airborne radioactivity monitoring instrumentation, including in the PSAR the criteria for placement and in the FSAR additional details as called for in Section 12.3.4, "Standard Format and Content...," for normal operation, anticipated operational occurrences, and accident conditions.

b. The criteria and method for obtaining representative in-plant airborne radioactivity concentrations (PSAR and update in FSAR).

c. Information describing the implementation of Regulatory Guides 1.21, 8.2, 8.8, 1.97 and ANSI N13.1. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.3 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. Specific acceptance criteria for these areas of review are as follows:

1. FACILITY DESIGN FEATURES

Acceptability of the facility design features will be based on evidence that the applicant has applied the guidance in Regulatory Guide 8.8 or that alternatives have been proposed. This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling, processing, etc., in-service inspection and calibration) have been considered in plant design and that potential radiation exposure from these activities will be kept ALARA by radiation protection features incorporated in the design, including ease of accessibility to work and inspection and sampling areas, the ability to reduce source intensity, design measures to reduce the production, distribution, and retention of activated corrosion products, the ability to reduce time required in radiation fields, provision for portable shielding and remote handling tools, etc. Access control will be judged for acceptability in accordance with the requirements of 10 CFR §20.303.

The areas inside the plant structures, as well as the general plant yard, must be identified by showing radiation areas with acceptable maximum design dose rates and zones. Maximum zone dose rate should be defined for each zone, as well as anticipated occupancy and access control. Acceptance criteria are as follows: The areas that have to be occupied on a predictable basis (number of people and stay or transit times) during normal operations and anticipated operational occurrences (including refueling; purging; fuel handling and storage; radioactive material handling; processing, use, storage and disposal; normal maintenance; routine operational surveillance; inservice inspection; and calibration) should be zoned such that this occupancy results in an annual dose to each of the involved individuals that is below the limits of 10 CFR Part 20 and is as low as is reasonably achievable, and a total man-rem dose that is as low as is reasonably achievable. Based on current operating experience and on predictions made for new plant designs, it is expected that the plant shielding can be designed, the plant can be zoned and sufficient radiation protection design features can be incorporated such that these individuals would receive a small fraction of the 10 CFR Part 20 limit. Whether radiation protection design and zoning is acceptable will be based partly on the actual numbers for average annual radiation exposure to these individuals, determined in the dose assessment required in Section 12.4.

2. SHIELDING

The shielding design is evaluated as to the assumptions used to calculate shield thickness, the calculational method used, and the parameters chosen. There are a number of acceptable shielding calculational codes available for use that are effective for determining the necessary shield thickness for gamma ray sources and for combination neutron-gamma sources. Most of the codes used by shield designers have been entered into the code description file of the Radiation Shielding Information Center at Oak Ridge National Laboratory, which means they have been tested and authenticated for operation but not for reliability and accuracy. RAB has three codes in-house for use in shielding calculations. These are SDC, a kernal integration shield design code; G^3, a general purpose gamma ray scattering program; and MORSE, a general purpose Monte Carlo multigroup neutron and gamma ray transport code. SDC can calculate gamma ray shielding requirements, handling 12 source geometries (including point, line, disk, plane, slab, and sphere) and with cross sections and materials compositions for 17 materials. As many as 12 gamma ray energy groups, covering the range from 0.1 to 10 MeV, may be used to describe the gamma ray spectrum. The staff will use these codes, as necessary, to calculate dose rates for given shield designs and source strengths, as a confirmation of the applicant's method.

The applicant's shielding design is acceptable if the methods are comparable to commonly acceptable shielding calculations and assumptions regarding source terms, cross sections, shield and source geometries, and transport methods are realistic. Acceptable shielding codes include but are not limited to ANISN, DOT, MORSE, SAM-CE, 05R, 06R, G^3, SDC, and many others. This listing does not imply that all these codes are equivalent, since some are much more sophisticated than others. The staff believes it is advantageous to use a good calculational procedure, since an effective shield design is essential to meeting the criteria that occupational radiation exposures will be as low as is reasonably achievable.

Two documents provide additional guidance for acceptability of the shielding design. One is "Reactor Shielding for Nuclear Engineers," edited by N. M. Schaeffer, published by AEC-OIS, 1973." The second is the Stone & Webster Engineering Corporation topical report RP-8 entitled "Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants." These documents provide useful guidance regarding radiation shielding design. Some limitations are noted for RP-8, in that the labyrinth entrance ways may not provide dose rates at the outside entrance consistent with area radiation zoning.

In addition, Regulatory Guide 1.69 provides guidance on the fabrication and installation of concrete radiation shields for nuclear power plants. Acceptability of the shield construction will be based on an indication that the guidance of this document has been implemented in the facility construction, or that acceptable alternatives have been proposed. Regulatory Guide 8.8 provides additional acceptance criteria regarding shielding and isolation in radiation protection design.

3. VENTILATION

The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the areas covered in SAR Section 12.2.2 will assure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents, and that the concentrations of radioactive material in areas normally occupied can be maintained in accordance with the requirements of 10 CFR Part 20. The system shall have adequate capability to reduce concentrations of airborne radioactivity in areas not normally occupied where maintenance or in-service inspection has to be performed, to levels in accordance with the requirements of 10 CFR §20.103. The system shall be designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel in normally occupied areas. Acceptability of the ventilation system, relative to radioactive gases and particulates will also be based on evidence that the applicant has applied the guidance of Regulatory Guide 8.8 or that alternatives have been prepared.

Regulatory Guide 1.52, particularly Sections C.4 and 5, provides guidance that can be used in this review, although the guide is written with regard to mitigating accidents involving airborne radioactivity. Good practice in that regard is applicable to normal operation as well, since release of radioactivity in normal operational occurrences is usually different only in quantity from some of the accident cases.

4. AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING SYSTEMS

The area radiation monitoring systems will be acceptable if they meet the following criteria:

1. The detectors are located in areas which may be normally occupied without restricted access and which may have a potential for radiation fields in excess of the radiation zone designation discussed in Subsection 12.3.II.1.

2. The detectors provide on-scale readings of dose rates that include the design maximum dose rate of the radiation zone in which they are located as well as the maximum dose rate for anticipated operational occurrences.

3. The detectors are calibrated annually or during fuel outages and after any maintenance work is performed on the detector.

4. Each monitor has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms.

5. There is readout and annunciation in the control room.

b. The airborne radioactivity monitoring system will be acceptable if it meets the following criteria:

1. Air should be sampled at normally occupied locations where airborne radioactivity may exist, such as solid waste handling areas, spent fuel pools, reactor operating floors, and BWR turbine buildings. The monitoring system should be capable of detecting ten MPC-hours of particulate and iodine radioactivity in any compartment which has a possibility of containing airborne radioactivity and which may be occupied by personnel. Continuous monitoring of air being exhausted from locations within the facility during normal operation is an acceptable method. The use of noble gas monitors to infer airborne radioiodine alarm levels is also an acceptable method.

2. Representative air concentrations are measured at the detectors, which are located as close to the sampler intakes as possible.

3. Ventilation monitors are upstream of HEPA filters.

4. The detectors are calibrated routinely and after any maintenance work is performed on the detector.

5. Each location has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms.

6. There is readout and annunciation in the control room.

c. The in-plant accident radiation monitoring systems should provide personnel with the capability to assess the radiation hazard in areas which may be accessed during the course of an accident. The accident montoring systems may include the normal area radiation monitors, airborne radioactivity monitors, and portable radiation

measurement equipment. Portable instruments which are to be used in the event of an accident should be placed to be readily available to personnel responding to an emergency. Emergency power should be provided for installed accident monitoring systems. The accident monitoring systems should have usable ranges which include the maximum calculated accident levels, and they should be designed to operate properly in the environment caused by the accident.

Instrumentation designated to determine very high radiation fields in the event of an accident may be placed such that built-in or added shielding provides protection to instruments from radiation damage. In such cases these instruments will need to be calibrated for the accident conditions. These calibrations however may take advantage of calculated shielding factors, using appropriate gamma energy spectrum distributions. The range of these instruments must be adequate to monitor the maximum radiation fields calculated in the event of an accident.

d. Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants," provides useful guidance that is applicable to the acceptability of airborne radioactivity monitoring in-plant. Regulatory Guide 8.2 includes guidance on surveys to evaluate radiation hazards. American National Standard ANSI N13.1 provides detailed guidance on sampling airborne radioactive materials in nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. Regulatory Guide 8.8 provides guidance on monitoring systems.

e. Instrumentation to monitor for accident criticality will be acceptable if it meets the criteria of 10 CFR Part 70.24 (a)(1), and Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

III. REVIEW PROCEDURES

The information on radiation protection design features furnished in the SAR, including referenced parts of Chapters 9 and 11, is reviewed for completeness in accordance with the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The reviewer evaluates the SAR text and the scaled layout drawings of the facility, concentrating on the sources, shielding, and layouts for the auxiliary building, including the radwaste systems, decontamination facilities, office and access control areas, laundry, lockers and shower rooms, and laboratory facilities; the fuel handling facilities, including the spent fuel pool and related equipment; and the BWR turbine building, including location of steam lines, reheaters, and moisture separators. For the PSAR this review is particularly concerned with preliminary design features which appear to be contradictory to assuring that ORE will be ALARA. In this review, radiation protection design features are evaluated using the guidelines of Regulatory Guide 8.8. The access control plans are reviewed both to determine conformance with 10 CFR Part 20 and to determine whether they will control access properly in limited access areas and in restricted access areas (high radiation areas). The reviewer examines locations of critical controls, valve operating stations, pumps, sample collection stations, inservice inspection locations, radiation monitors, control panels, major pipes carrying radioactivity, filters for radioactive liquids and gases, and unshielded low level radioactive material storage or

processing tanks. SAR Chapters 9 and 11 are reviewed to cover specific details of the fuel handling and storage systems, ventilation systems, and radwaste systems as they relate to radiation protection design. Chapter 9 will provide the major description of the mechanical features of ventilation systems with regard to the venting airborne radioactivity from the plant. Chapter 9 will also cover major features of the spent fuel pool design, the fuel handling system design and the spent fuel pool cleanup system. Chapter 11 may cover some of the design details of gaseous, liquid and solid radwaste systems that relate to radiation protection. The reviewer evaluates all relevant aspects of the initial design plans particularly to identify new arrangements, improved designs, unusual shield thicknesses, a new or modified shield thickness calculational procedure, unusual assumptions in the calculation, placement of radiation monitors, etc.

RAB evaluates the adequacy of the applicant's shielding design on the basis of acceptable radiation shielding codes. RAB makes a verifying check calculation with SDC, G^3, or MORSE, whichever is specifically applicable to the situation.

For the FSAR the reviewer considers any changes in the design that might necessitate changes in operating procedures to accommodate a changed radiation zone or a different location of equipment.

The reviewer determines whether the applicant has followed the guidance of the referenced regulatory guides and industry standards, both by comparison of the applicant's methods with the information in the guides and by the applicant's reference to any such guides or to alternatives that have been proposed. The reviewer evaluates whether the alternatives are equivalent to or improvements on the methods cited in the referenced regulatory guides. Alternatives that are neither of these are likely to be disapproved.

Based on the review, RAB may request additional information or request the applicant to reevaluate the radiation protection design features to meet the acceptance criteria of Subsection II.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's Safety Evaluation Report. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

The radiation protection design features at (plant name) are intended to help maintain occupational radiation exposures as low as is reasonably achievable. Many of these design features have been incorporated as a result of the applicant's radiation protection design review and from radiation exposure experience gained during the operation of other nuclear power plants. (Include examples of design features that have been incorporated into the design to reduce the need for maintenance and other operations in radiation fields; reduce radiation sources where operations must be performed; allow quick entry and easy access; provide remote operation capability or reduce the time required for work in radiation fields;

and examples of other features that reduce radiation exposure of personnel.) These design features are consistent with those contained in Regulatory Guide 8.8 and are acceptable.

Areas within the restricted area will be divided into (number of zones) radiation zones. The dose rate criterion for each of these zones is derived from expected occupancy and access restrictions. These criteria are then used as the basis for the radiation shielding design. This allows for arrangements of radioactive equipment that are in accordance with the requirements of 10 CFR Part 20 and the guidelines of Regulatory Guide 8.8. During plant operation and refueling conditions, the Health Physics staff will evaluate area access classifications and monitor entry into areas to update posting and entry requirements for 10 CFR Part 20.203.

The radiation shielding will be designed to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public. The following are several of the shielding design features incorporated into (plant name). (List several examples of shielding design features used at plant.) Some of the criteria used by (utility) in locating penetrations in shield walls at (plant name) are: (list several shield penetration location criteria used.) These shielding techniques are designed to maintain personnel radiation exposures as low as is reasonably achievable and are acceptable.

The general shield design methodology and source term inventories used for (plant name) are similar to those described in the topical report (topical report name if applicable), which has been previously reviewed and accepted by the staff. The basic radiation transport analysis used for the applicants' shield design is based on (list appropriate shielding computer codes used). The applicant also used shielding information from operating nuclear plants as input data for their shield design calculations. All concrete shielding in the plant will be constructed in general compliance with Regulatory Guide 1.69. We find the shielding design and methodology presented in the (Preliminary or Final) Safety Analysis Report acceptable based on the criteria of the Standard Review Plan.

The ventilation system at (plant name) will be designed to protect personnel and equipment from extreme thermal environmental conditions and ensure that plant personnel and the general public are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR Parts 20 and 50. The applicant intends to maintain personnel exposures as low as is reasonably achievable by: 1) maintaining air flow from areas of potentially low airborne contamination to areas of higher potential concentrations; 2) ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants, respectively; and 3) locating ventilation system intakes so that intake of potentially contaminated air from other building exhaust points is minimized. These design criteria are in accordance with the guidelines of Regulatory Guide 1.52 and Regulatory Guide 8.8. (List examples of exposure reduction features in the ventilation system.)

The applicant's area radiation monitoring system is designed to: 1) monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present; 2) alarm when the radiation levels exceed preset levels to warn of excessive radiation levels; and 3) provide a continuous record of radiation levels at key locations

throughout the plant. In order to meet these objectives, the applicant plans to use (number) area monitors located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system is equipped with local and remote audio and visual alarms and a facility for central recording. (List examples of other area monitoring system features.)

The design objectives of the applicants' airborne radioactivity monitoring system are: 1) to assist in maintaining occupational radiation exposure to airborne contaminants as low as is reasonably achievable; 2) to check on the integrity of systems containing radioactivity which are being monitored; and 3) to warn of inadvertent release of airborne radioactivity to prevent overexposure of personnel. The applicant will install airborne radioactivity monitors in work areas where there is a potential for airborne radioactivity. These airborne radioactivity monitors have the capability to detect ten MPC-hours of particulate and iodine radioactivity in any compartment which has a possibility of containing airborne radioactivity and which may be occupied by personnel. The applicant will provide portable continuous air monitors when needed to monitor air in areas not provided with fixed airborne radioactivity monitors. All airborne and area radioactivity monitors will be calibrated at ( ) month intervals. (List examples of other airborne radioactivity monitoring features.) The objectives and location criteria of (plant name's) area and airborne radiation monitoring systems are in conformance with 10 CFR Parts 20 and 50 and Regulatory Guides 8.2 and 8.8.

The objective of the applicant's accident radiation monitoring system is to provide the capability to assess the radiation hazard in areas which may be accessed during the course of an accident. The installed instruments have emergency power supplies; and the portable instruments are placed to be readily accessible to personnel responding to an emergency. The systems are designed for use in the event of an accident in terms of 1) usable instrument range, and 2) the environment the instrument can withstand.

V. REFERENCES

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

2. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable."

3. ANSI N13.1 "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," American National Standards Institute.

4. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

5. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."

6. "Reactor Shielding for Nuclear Engineers," N. M. Schaeffer, Editor; published by USAEC-OIS, 1973.

7. Regulatory Guide 1.52, "Design, Test, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

8. Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

9. 10 CFR Part 20, "Standards for Protection Against Radiation."

10. Stone and Webster Topical Report, "Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants," RP-8, 1974.

11. Regulation Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

12. Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

13. ANSI N16.2, "Criticality Accident Alarm Systems."

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SECTION 12.4 DOSE ASSESSMENT

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) relating to in-plant and onsite radiation dose assessment are reviewed:

1. The expected occupancy of plant radiation areas, including numbers of personnel and times of occupancy. The estimated annual occupancy for each radiation zone and the dose rates at the points of occupancy and basis for the occupancy and the dose rate values (preliminary safety analysis report, PSAR and update in final safety analysis report, FSAR).

2. The objectives and criteria for design dose rates in-plant and at onsite areas (PSAR and update in FSAR).

3. The estimated annual man-rem doses associated with major functions such as operation, radwaste handling, normal maintenance, refueling, and in-service inspection and the average individual radiation exposure resulting from these activities (PSAR and update in FSAR).

4. The estimated annual dose at the boundary of the restricted area and to construction workers at a multi-unit plant due to radiation from onsite sources (PSAR and update in FSAR).

5. The description of any measures taken as a result of the dose assessment process presented in this section and/or of the design review process in Section 12.1 to reduce particular estimated man-rem doses for specific functions or activities (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Power Plants - Design Stage Man-Rem Estimates." The dose limits in 10 CFR §20.101 for exposure of individuals to radiation in restricted areas,

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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.

and in 20.103 for exposure of individuals to concentrations of radioactive materials in restricted areas are upper limits and the plant must be designed so that they are not exceeded. The dose assessment is a key factor in determining if the plant design and proposed methods of operation assure that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). Acceptability will be based on the thoroughness with which the applicant displays and demonstrates that occupancy factors, dose rates in various occupied areas, and number of personnel needing to be involved are evaluated for the various areas of the plant, as specifically designed, and for the various functions that will be carried out. Occupancy factors will be acceptable if it is demonstrated that they are based on operating experience, coupled with the applicant's plan for operating the proposed plant, or that they are based on a thorough analysis of the future plant operation. Estimates should be included for special high dose accumulating operations such as normal maintenance, radioactive material handling, refueling, and in-service inspection. All assumptions used in the assessment should be included.

The applicant should provide realistic estimates of doses resulting from nonroutine or special maintenance activities (e.g., steam generator tube plugging), explaining the use of available data on dose rates, occupancy times, and probability of occurrence. All such exposures, averaged over the life of the plant, should be included in the total, whether the exposed individuals are plant, utility, contractor, or labor contract worker.

The applicant should describe the dose assessment process in sufficient detail to make it clear that the process has involved a review of each activity potentially involving significant occupational radiation exposure. Specific examples of any action taken should be provided.

Dose rates in areas for which occupancies are given will be acceptable if they are based on generally used calculational procedures with realistic source, attenuation, and distance factors.

Estimates of the number of personnel involved in various operations and various areas of the plant will be acceptable if they make realistic assumptions on use of the plant work force and include input from experience at operating reactors. The sources of these data should be cited.

All applicants are to demonstrate that the designs and the operating plans include reasonable efforts to assure that occupational radiation exposures will be ALARA. In addition, applicants for licenses for boiling water reactor (BWR) plants are to demonstrate that they have made reasonable efforts to assure that occupational radiation exposures due to N-16 sources in turbine buildings will be ALARA. For BWR sites expected to contain more than one unit, where one or more units may be operating while others are under construction, the applicant is to provide an analysis to demonstrate that he has made reasonable efforts to assure that onsite population exposures due to such sources will be ALARA, including that to the expected construction force. The requisite analysis will be acceptable if the applicant demonstrates that he has appropriately considered the following factors and their interactions:

a. The number of reactor units, their power levels, and their expected rates of completion.
b. The relative orientations of pertinent structures and their expected completion times.
c. The overall sizes and occupational compositions of the construction force and of the plant work force.
d. The locations and orientations of components with significant inventories of N-16 with respect to structural concrete.
e. The need for additional shielding for significant source-containing components.

The staff's policy on acceptance criteria for average annual radiation exposure to plant operating personnel during predictable activities has been stated in Section 12.3 above, in the discussion on acceptable criteria for plant radiation zoning. Also involved in the acceptance criteria for the dose assessment is the applicant's description of design and operating features for increasing accessibility to work, inspection, and sampling areas, for reducing the intensity of radiation sources that have to be worked around, for reducing the production, distribution, and retention of activated corrosion products, for reducing the time required for work in radiation fields, and for providing additional methods for reducing occupational radiation exposure. Judgment of how realistic the dose assessment is and how appropriate the ALARA operating and design features are will be used in determination of acceptability of the average annual occupational radiation exposure estimates. However, numerical acceptance criteria for total annual man-rems resulting from plant operation have not been developed. Nevertheless, the value obtained in the assessment made in this section provides a basis for judgment of the radiation protection program and a determination if ORE will be ALARA.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

The reviewer evaluates the text and scaled layout drawings of the facility to determine the manner in which the plant is zoned for radiation dose rate and to evaluate proposed occupancy. The reviewer checks the estimated occupancy factors, the calculational procedures, the dose rate values, and the in-plant airborne radioactivity concentrations. The reviewer compares the rationale and assumptions used in the dose assessment with those provided for other accepted plants and with operating experience and referenced methods. The reviewer determines whether the applicant's evaluation is thorough and realistic. The reviewer forms a judgment on whether the annual occupational radiation exposure man-rem estimate is ALARA. Under circumstances where the reviewer decided the value was not ALARA, he could request design or procedures improvements. Based on the review, RAB may also request additional information or request the applicant to modify or improve the analysis for the purpose of meeting the acceptance criteria given in Subsection II.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's Safety Evaluation Report. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings.

The applicant has based the estimate of annual man-rem exposure at (plant name) on plant specific projections as to occupancy and dose rate, on experience from currently operating reactors of other utilities, their own operating experience, and improvements in design of systems to maintain in-plant radiation levels as low as is reasonably achievable. (Give examples of ALARA design features used to reduce man-rems.)

The applicant has performed an assessment of the doses that will be received by plant and contractor personnel. This dose assessment is based upon occupancy factors, expected dose rates, expected airborne radioactivity concentrations, and estimates of the time and manpower necessary to perform the various tasks involved in plant operation. The dose assessment includes a breakdown of the annual man-rem doses associated with major functions such as normal operations, and maintenance, radwaste handling, refueling, and inservice inspection. The applicant estimates the total annual collective rem dose to plant personnel and contractors to be ( ) man-rems. This estimate is based on experience from operating plants and on information presented in (list sources used, e.g., NUREG documents, articles, etc.). Currently operating light water reactors average ( ) man-rems per unit annually. (Compare utilities man-rem estimate with current average man-rems. Discuss contractor exposure.) We find the bases for (plant name's) exposure estimate acceptable, and consistent with the acceptable criteria in the Standard Review Plan.

(Utility) provides a tabulation of the maximum expected radioactive airborne concentrations, as well as estimates of the inhalation and submersion dose equivalent rates to plant personnel. The dose equivalent rates are derived from the airborne radioactivity source terms given in Chapter 11. The applicant's assumptions and models upon which his internal and submersion dose estimates are based for occupational exposures are consistent with the staff's and are acceptable.

V. REFERENCES

1. ICRP Publication 22, "Implications of Commission Recommendations That Doses Be Kept As Low As Readily Achievable," International Commission on Radiation Protection (1973).

2. L. A. Johnson, "Occupational Radiation Exposure at Light Water Cooled Power Reactors," 1976.

3. 10 CFR Part 20, "Standards for Protection Against Radiation."

4. R. Wilson, "Man-rem Economics and Risk in the Nuclear Power Industry," NUCLEAR NEWS, February 1972.

5. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

6. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Power Plants - Design Stage Man-Rem Estimates."

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SECTION 12.5 HEALTH PHYSICS PROGRAM

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) related to the health physics program are reviewed as part of the radiation protection program.

1. ORGANIZATION
   a. The administrative organization of the health physics program, including the authority and responsibility of each position identified (preliminary safety analysis report, PSAR and update in the final safety analysis report, FSAR).
   b. The experience and qualifications of the personnel responsible for the health physics program and for handling and monitoring radioactive material. Reference may be made to SAR Chapter 13 as appropriate (final safety analysis report, FSAR).
   c. Information describing the implementation of Regulatory Guides 8.8, 8.10, 8.2 and 1.8. Information describing alternatives, if such are proposed (PSAR and update in FSAR).
   d. Review of qualifications, experience, and organization will be coordinated with QAB.

2. EQUIPMENT, INSTRUMENTATION, AND FACILITIES
   a. The criteria for selecting portable and laboratory technical equipment and instrumentation for performing radiation and contamination surveys, for in-plant airborne radioactivity monitoring and sampling, for area radiation monitoring, and for personnel monitoring for normal operation, anticipated operational occurrences and accident conditions (PSAR and update in FSAR).
   b. The description of instrument storage, calibration, and maintenance facilities (PSAR and update in FSAR).
   c. The description and location of the health physics facilities (including locker and shower rooms, respiratory protective equipment, and other contamination control equipment and areas, and information describing how such facilities and services will allow male and female workers to receive the necessary protection against radioactive contamination (PSAR and update in FSAR).

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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.

d. The location of items in 2a, b, and c and the description of types of detectors and monitors, sensitivity, range, and frequency and methods of calibration (FSAR).

e. Information describing the implementation of Regulatory Guides 8.4, 8.8 and 8.9. Information describing alternatives, if such are proposed.

3. PROCEDURES

a. The description of physical and administrative measures for controlling access and stay time in radiation areas (FSAR).

b. The description of procedures and methods of operation for assuring that occupational radiation exposure (ORE) will be as low as is reasonably achievable (ALARA) (FSAR).

c. The description of methods, frequencies, and procedures for conducting radiation surveys (FSAR).

d. The description of the bases and methods for monitoring and control of personnel and equipment and of surface contamination including reporting practices (FSAR).

e. The description of methods and procedures for evaluating and controlling potential airborne radioactivity concentrations, for special air sampling and the issue and use of respiratory equipment, and for handling and storage of sealed and unsealed byproduct, source and special nuclear materials (FSAR).

f. The description of radiation protection training programs (FSAR).

g. Information describing the implementation of Regulatory Guides 8.8, 8.10, 8.2, 8.7, 8.9, 1.16, 1.39, and 1.8. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.5 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

Specific acceptance criteria for the areas of review given above are as follows:

1. ORGANIZATION

Acceptance will be based on a determination that the organization described, along with the duties, qualifications, and training of the individuals responsible for assuring that ORE will be ALARA are in accordance with Regulatory Guides 8.8, 8.10, 8.2, and 1.8. Alternatives will be evaluated on the basis of a comparison with the referenced Regulatory Guides.

2. EQUIPMENT, INSTRUMENTATION, AND FACILITIES

The following shall be included in order that these items are acceptable:

a. The radiochemistry laboratory is equipped to perform routine analyses required for personnel protection, surveys, and related health physics functions.

b. The counting room (low background) is equipped and has the necessary instrumentation to perform routine counting on all plant radioactivity samples (water, air, swipe survey, etc.) in conformance with 10 CFR Part 20. Counting room equipment to normally include the following:

(1) Multi-channel gamma pulse height analyzer.

(2) Low background alpha-beta proportional counter and gamma ray and alpha-beta scintillation counters. Regulatory Guide 5.9 provides specifications for Ge(Li) spectroscopy systems that may be useful for application to the gamma ray system.

(3) End window G-M type counter.

c. Portable instruments for measuring radiation or radioactivity to include:

(1) Low and high range ion chamber rate meter type instruments.

(2) Portable G-M detectors.

(3) Alpha scintillation or proportional counter rate meters.

(4) Neutron dose rate detector.

(5) Air samplers for use with particulate filters and iodine collection devices (such as charcoal cartridges) and airborne radioactivity monitors.

d. Personnel monitoring instruments to include:

(1) Friskers for detecting radioactive contamination.

(2) Self-reading low and intermediate pocket dosimeters (for early evaluation of individual doses). Performance and other requirements shall conform to Regulatory Guides 8.4 and 8.14, or to appropriate proposed alternatives.

(3) Film badges and/or thermoluminescent dosimeters (TLD).

(4) Provisions for bioassay and whole body counting to meet the requirements of 10 CFR Part 20 and Regulatory Guide 8.9, or to appropriate proposed alternatives.

e. Utility-issued personnel protection equipment to included:

(1) Anti-contamination clothing.

(2) Plastic suits for liquid contamination control.

(3) Head covers, shoe covers, gloves, and safety related items.

(4) Pressure/Demand full-face-piece air line respirators.

(5) Continuous air flow two-piece plastic suits for covering whole body.

(6) Pressure demand full-face-piece self-contained breathing apparatus.

(7) Full-face mechanical filter respirators.

f. Personnel protective clothing and equipment that meet the requirements of the American Standards Institute ANSI Z88.2 and the U.S. Bureau of Mines approved schedules for use in atmospheres containing radioactive materials.

g. As a minimum the following health physics support facilities or areas be provided:

(1) Portable instrument calibration and storage area. The latter should be easily accessible.

(2) Personnel decontamination area with necessary monitoring equipment. This facility should be located and designed to expedite rapid cleanup of personnel and should not be used as multiple purpose area.

(3) Facility and equipment to clean, sanitize, repair, and decontaminate personnel protective equipment, monitoring instruments, respirators, etc.

(4) A change room.

(5) Control points for entrance or exit into controlled access areas of the plant.

(6) One or more health physics stations, which may be used as the location for portable radiation survey equipment, respiratory protective equipment, personnel monitoring equipment, and contamination control supplies. These stations and the equipment should be readily accessible and equipped to facilitate communication throughout the plant.

Acceptance will also be based on implementation of the guidance of Regulatory Guide 8.8 or the provision of acceptable alternatives.

3. PROCEDURES

Plans and procedures will be acceptable if they meet the criteria provided in Regulatory Guides 8.8, 8.10, 1.33, and 8.2 or proposed appropriate alternatives. There should be provisions for a special control procedure for any zone 4 or higher area that includes a special survey of the area before entry and the development of a radiation work permit program. The work permit program should include the following: data on radiation levels in the area before entry and the development of a radiation work permit program. The work permit program should include the following: data on radiation levels in the area, allowable working time, protective clothing and respiratory protective equipment, special tools, portable shielding, and health physics and special personnel monitoring devices. Maintenance, repair, surveillance, and refueling procedures and methods used by the applicant should be reviewed to assure that occupational radiation exposures will be ALARA and in accordance with Regulatory Guide 8.8. For major dose accumulating functions, a post-operation review should be conducted to evaluate the effectiveness of the work permit program in assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). There should be provisions for supervision and control of the handling or movement of material within and from radiation or controlled access areas. Acceptance criteria for contamination control limits are being developed. There shall also be provisions for personnel monitoring procedures, bioassay, keeping records of personnel doses, and the reporting of personnel doses. 10 CFR §20.102, .201, .401, and .407 provide the criteria for radiation surveys, personnel monitoring, bioassay, record keeping, and reporting. Guidance regarding these areas is provided by Regulatory Guide 8.2 (surveys and personnel monitoring), 8.3 (personnel monitoring equipment), 8.9 (bioassay), and 1.16, 8.2, 8.7 (record keeping and reporting), 8.8 (decontamination, inspection, radiation protection program, and operations), 8.13 (training on radioactive risks to fetuses), and 8.14 (neutron personnel monitoring).

The acceptability of the health physics program will also be based on provisions for the indoctrination and personnel training and retraining programs. Regulatory Guides 8.8, 8.10, and 1.8 provide information regarding these areas. Section 19.12 of 10 CFR Part 19 requires instruction of personnel on radiation protection. There should be a regular review of the radiation protection program, which should include updating procedures, equipment, and facilities where improvements are possible. The program should include regular audits to determine where occupational radiation exposures are occurring and to review possible methods for reducing these exposures.

With regard to plant cleanliness, which is critical where radioactive material is concerned, Regulatory Guide 1.39 discusses housekeeping requirements that are applicable to operation as well as construction. Finally 10 CFR Part 70 provides the guidance on special nuclear, source, and byproduct materials.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

RAB reviews all the areas discussed in I., evaluating acceptability by making the referenced comparisons with regulations, regulatory guides, and industry standards. These can be summarized as follows:

1. The organizational position, functional responsibilities, experience, and qualifications of persons responsible for the health physics program. The plant organization, the functional responsibilities, and the qualifications of personnel are the primary responsibility of the Quality Assurance Branch, and are reviewed as part of Chapter 13. RAB reviews the health physics/radiation protection function and personnel qualifications, in accordance with Regulatory Guide 1.8.

2. The equipment necessary to measure radioactivity and radiation fields and exposures, including the number, type, range, sensitivity, calibration method and frequency, availability, and planned used of portable, fixed, laboratory, and personnel monitoring instrumentation.

3. The health physics facilities and associated protective equipment for controlling ORE and contamination.

4. The training and indoctrination program and health physics instruction manuals, as well as the respiratory protective equipment fitting program. Plant procedures are the primary responsibility of the Operator Licensing Branch which reviews in Chapter 13 (Section 13.5), such formal procedures as the plant radiation protection procedures. RAB reviewers of health physics/radiation protection procedure commitments should communicate any problems to the OLB.

5. The procedures to control storage and movement of radioactive material, to control exposures, and to control contamination. Where these procedures are part of the formal plant operating procedures, the review will include informing the OLB of any problems as in 4 above.

Based on the review, RAB may request additional information or request the applicant to modify his submission in order to meet the acceptance criteria described in Subsection II.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's Safety Evaluation report. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

The health physics program objectives are to provide reasonable assurance that the limits of 10 CFR Part 20 are not exceeded, to further reduce unavoidable exposures, and to ensure that individual and total man-rem occupational radiation exposures are maintained as low as is reasonably achievable.

The duties of the plant (Health Physicist) include (list duties). The health physics organizations, the qualifications of the health physics personnel, the objectives of the Health Physics program, and the ways in which it will be implemented are in accordance with the guidelines of Regulatory Guides 8.8, 8.2, 8.10, and 1.8 and are acceptable.

The radiation protection features at (plant name) include a (health physics laboratory radiochemistry lab, personnel decontamination and emergency treatment areas, an access control point, counting room, calibration room, respirator testing facility, health physics office, laundry, etc.). These facilities are sufficient to maintain occupational radiation exposures as low as is reasonably achievable and are consistent with the provisions of Regulatory Guide 8.8.

Equipment to be used for radiation protection purposes includes (portable radiation survey instruments, personnel monitoring equipment, fixed and portable area and airborne radioactivity monitors, laboratory equipment, air samplers, respiratory protective equipment, protective clothing, etc.). The number and types of equipment to be used is adequate and provides reasonable assurance that the applicant will be able to maintain occupational exposures as low as is reasonably achievable.

All permanent and temporary plant personnel will be assigned (beta-gamma thermoluminescent dosimeter badges or film badges) to be worn at all times. These badges will be processed (monthly), or more frequently if significant exposures are suspected. All personnel assigned (TLD or film badges) are also required to wear (direct or indirect) reading dosimeters when entering the controlled area. The readings from these dosimeters will be used to keep a running total of an individual's dose prior to TLD or film badge processing. Plant visitors wear self-reading dosimeters or are escorted by an individual wearing such personnel dosimetry devices. Neutron film badges, neutron dosimeters, and alarming dosimeters will also be provided for personnel when necessary. Whole body counts of all plant personnel will be conducted on a scheduled basis and other bioassays will be provided when deemed necessary by the (Health Physicist or Radiochemical Engineer). All radiation exposure information will be processed and recorded in accordance with 10 CFR Part 20.

Maintenance, repair, surveillance and refueling procedures and methods used by the applicant are reviewed to assure that all plant radiation protection procedures, practices, and

criteria have been considered, to assure that occupational radiation exposures will be ALARA and in accordance with Regulatory Guide 8.8. Procedures are also developed to assure that exposure limits are not exceeded by plant or visitor personnel onsite; to administer and control conditions of radiation work permits; to post radiation areas; to establish radiation access control zones; to control all radioactive material entering or leaving the plant site; and to train plant and visitor personnel in radiation protection policies and procedures and meet the requirements of Regulatory Guide 1.33.

Based on the information presented in the (PSAR, FSAR) and the applicant's responses to our questions, we conclude that the applicant intends to implement a radiation protection program that will maintain in-plant radiation exposures within the applicable limits of 10 CFR Part 20 and 50 and will maintain exposures as low as is reasonably achievable.

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation."

2. 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections."

3. Regulatory Guide 1.8, "Personnel Selection and Training."

4. ANSI N18.7 "Administrative Controls for Nuclear Power Plants," American National Standards Institute

5. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

6. Regulatory Guide 8.6, "Standard Test Procedures for G-M Counters."

7. ANSI Draft N547, "Control of Radioactive Surface Contamination on Material, Equipment and Facilities to be Released for Uncontrolled Use." (in preparation)

8. Regulatory Guide 1.39, "Housekeeping Requirements for Water Cooled Nuclear Power Plants."

9. USBM-23, "Respiratory Protective Services for Use in Atmospheres Containing Radioactive Materials," U.S. Bureau of Mines (1973).

10. Regulatory Guide 8.7, "Occupational Radiation Exposure Records System."

11. Regulatory Guide 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters."

12. Regulatory Guide 8.3, "Film Badge Performance Criteria."

13. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."

14. ANSI Z88.2, "Procedures for Respiratory Protection," American National Standards Institute.

15. 10 CFR Part 20, Appendix B, Table 1, "Concentrations in Air and Water Above Natural Background."

16. Regulatory Guide 1.16, "Reporting of Operating Information."

17. Regulatory Guide 1.70, "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants."

18. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equation, and Assumptions for a Bioassay Program."

19. 10 CFR Part 70, "Special Nuclear Material."

20. Regulatory Guide 5.9, "Specifications of Ge(Li) Spectroscopy Systems for Natural Protection Measurements - Part I: Data Acquisition."

21. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

22. Regulatory Guide 8.14, "Personnel Neutron Dosimeters."

23. Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

24. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

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SECTION 13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

1. AREAS OF REVIEW

The QAB will review the management and technical organization of the applicant (and applicants for a manufacturing license, or for a standardized design approval) and its major contractors, including the NSSS vendor, and architect/engineer for the project. These organizations are typically located offsite, in contrast to the onsite plant operating organization which is reviewed under SRP Section 13.1.2. The review of offsite organizations includes their responsibilities and technical capability to engage in the activities proposed in the application. These activities include items such as reactor design, facility design, design review, design approval, manufacturing, construction management, testing and operation of the facility.

A. The PSARs, SSARs, and "Design Report" for the manufacturing license (see Appendix M of Part 50) should provide the following information:

The applicant's past experience in the design and construction of nuclear power plants, and past experience in activities of similar scope and complexity should be described. A description of the applicant's (utilities) management, engineering, and technical support organization, including organizational charts reflecting the applicant's current headquarters and engineering structure and planned modifications and additions to it to reflect the added functional responsibilities associated with the addition of the nuclear plant to the applicant's power generation capacity. These added responsibilities should be identified and should include items listed in 1 and 2 below. The description should include how these responsibilities are delegated and implemented within and from the headquarters staff and identify the working or performance level and responsible organization unit. Specific assignments for responsibility for the overall fire protection program and for associated functional level activities should be identified and described. The description should include an estimate of the number of persons expected to be assigned to each of the various units with responsibility for the project.

USNRC STANDARD REVIEW PLAN

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1. Design and Construction Responsibilities

These responsibilities are generally defined prior to submittal of the application to the Nuclear Regulatory Commission (NRC) and usually continue until the plant is turned over to the operating organization. A description of the implementation or delegation of the following areas of responsibilities should be included.

a. Principal site-related engineering work such as meteorology, geology, seismology, hydrology, demography, and environmental effects.

b. Design of plant and ancillary systems, including fire protection systems.

c. Review and approval of plant design features.

d. Site layout in respect to environmental effects and security provisions.

e. Development of safety analysis reports.

f. Material and components specification review and approval.

g. Procurement of materials and equipment.

h. Management of construction activities.

2. Technical Support for Operations

Technical services and backup support for the operating organization should become available in advance of the conduct of the preoperational and startup testing program and continue throughout the life of the plant. The special capabilities that should be included are:

a. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical and materials, and instrumentation and controls engineering.

b. Plant chemistry.

c. Health physics.

d. Fueling and refueling operations support.

e. Maintenance support.

f. Fire protection.

The PSAR should also identify general qualification requirements in terms of numbers, educational backgrounds, and experience for identified positions or classes of positions; and specific educational and experience background for assigned management and supervisory positions relative to Item 1 above.

For identified positions or classes of positions that have functional responsibilities for other than the identified application, the expected proportion of time assigned to the other activities should be described. In addition, the plans for providing technical support for the operation of the facility (Item 2 above) should be described.

The CP-stage review (PSAR) of the NSSS and A/E organizations (and applicants for a manufacturing license and the review of standardized designs) includes a review of their technical staff to perform the activity related to the application. The information submitted should include a description of the specific activity (including scope) to be engaged in, organizational description and charts reflecting their lines of authority and responsibility for the project, the number of persons assigned to the project, and qualification requirements for principal management positions related to the project. For those NSSS and A/E organizations with extensive experience, a detailed description of this experience may be provided in lieu of the details of their organization as evidence of technical capability.

B. The FSAR should provide the following information:

Organizational charts of the applicant showing the management and technical support headquarters structure, and a description of the specific provisions which have been made for technical support for operations (see A.2 above). The FSAR should (1) identify qualification requirements for headquarters staff personnel in terms of numbers, educational background and experience requirements for each identified position or class of positions providing headquarters technical support for operations, and (2) include specific educational and experience background for individuals holding the management and supervisory positions identified for Item A.2 above.

II. ACCEPTANCE CRITERIA

The applicant's description of its resources to deal with safety-related problems connected with the proposed addition of nuclear generating capacity should provide contributory evidence as to the technical qualifications of the applicant, as required by 10 CFR 50.34(a)(9) and (b)(7) or 10 CFR 50, Appendices M.5.e., N.2, and O.3. Criteria for acceptability include the following:

A. The applicant has identified and described the organizational groups responsible for implementing responsibilities for the project (CP) and/or operation of the facility (OL).

B. The applicant has described the method of implementing its responsibilities for dealing with the safety-related aspects of the project (CP) and operation of the facility (OL).

C. Clear management control and effective lines of authority and communications exist between (1) the organizational units involved in the design and construction of the project (CP); and (2) the organizational units involved in the management, operation, and technical support for the operation of the facility (OL).

D. Substantive breadth and level of experience and availability of manpower exist to implement the responsibility for the project and technical support for its operation. The need to supplement the corporate structure with additional experienced personnel for the initial years of operation will be determined on a case-by-case basis.

E. Qualifications of the "Engineer in Charge" should meet or exceed those given in Section 4.6.1 of ANSI N18.1, as endorsed by Regulatory Guide 1.8.

F. Responsibilities for fire protection should conform to BTP ASB 9.5-1.

III. REVIEW PROCEDURES

Each element of the SAR information is to be reviewed against this SRP section. The reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the magnitude and uniqueness of the project. Any exceptions or alternatives are to be carefully reviewed to assure that they are clearly defined and that an adequate basis exists for acceptance. The applicant should identify the revision of references, Regulatory Guides, and Codes and Standards used. The reviewer should identify the revision of references, Regulatory Guides, and Codes and Standards used in the review.

In the review and evaluation of the subject matter of this section of the SAR, the following points should be taken into consideration.

The management and technical support headquarters structure, as demonstrated by organizational charts and descriptions of functions and responsibilities, should be free of ambiguous assignments of primary responsibility. Design and construction responsibilities should be reasonably well defined in both numbers and experience of persons required to implement their responsibilities. The reviewer must recognize that there are many acceptable ways to define and delegate job responsibilities. At the CP stage, with respect to technical support for operations, the applicant's plans for headquarters staffing may not yet be firm. It is acceptable, therefore, if these plans are not fully specific in terms of numbers of people, provided the commitment made is sufficiently firm to assure the responsibility can be met. Variations in staffing may also be expected between applicants who lack prior experience with nuclear plant operation and those who have such experience. It is important that the reviewer assure himself that applicants in the former category do not underestimate the magnitude of the task. The reviewer

should be alert to the possibility that excessive workloads may be placed upon too small a number of individuals.

If the application involves the addition of more than one unit, the reviewer should assure that headquarters staffing plans take this fact into account. This is particularly important if additional units are scheduled to come on line at intervals of about one year or less, since the shakedown period for the operation of a new plant can be expected to produce quite heavy workloads. In some of these cases the applicant may plan to bolster the plant staff organization during such periods so that it is necessary to evaluate headquarters staffing plans in conjunction with those for the plant staff organization.

The reviewer should assess the degree of participation during the design and construction phases by that headquarters group that typically has plant operating (generating) responsibility. Interfaces between such a group and those with project engineering responsibilities should be examined.

The review procedure for this section consists, therefore, of:

1. An examination of the information submitted to determine that all subject matter identified in subsection I above has been addressed.

2. A comparison of the information with the acceptance criteria of subsection II above in the light of the additional points set forth earlier in this SRP section.

3. Review of information provided by the Office of Inspection and Enforcement position statement on the applicant's SAR organizational and administrative commitments.

In addition, if the applicant, as of the time the review takes place, has had experience in the operation of a previously licensed nuclear power plant, the reviewer may seek independent information relative to headquarters staffing and qualifications through the Office of Inspection and Enforcement, e.g., by discussion with inspection personnel or review of inspection reports.

The reviewer then determines, based on the foregoing, the overall acceptability of the applicant's management and technical support organization and staffing plans.

IV. EVALUATION FINDING

The reviewer verifies that the information presented and its review support conclusions of the following type to be used in the staff's safety evaluation report:

A. CP Safety Evaluation Report

"The applicant has described clear responsibilities and associated resources for the design and construction of the facility and has described its plans for management

of the project and for utilization of the NSSS and A/E. These plans have been reviewed and give adequate assurance tnat an acceptable organization and staff resources have been established to satisfy the applicant's commitments for the design and construction of the facility."

B. OL Safety Evaluation Report

"The applicant has described its organization for the management of, and its means for providing technical support for the plant staff during operation of the facility. These measures have been reviewed and we conclude that the applicant has an acceptable organization and adequate resources to provide offsite technical support for the operation of the facility."

V. REFERENCES

A. "Utility Staffing and Training for Nuclear Power," WASH-1130, Revised, U.S. Atomic Energy Commission, June 1973.

B. 10 CFR 50.34(a)(9) and (b)(7).

C. 10 CFR 50 Appendices M(5.), H(2.) and O(3.).

D. USNRC Branch Technical Position ASB 9.5-1.

E. Regulatory Guide 1.8, "Personnel Selection and Training."

VI. APPLICABILITY

The revisions shown in this section of the SRP will be implemented for all license applications submitted after May 1, 1979.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 13.1.2 OPERATING ORGANIZATION

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

The applicant's operating organization, as described in its Safety Analysis Report (SAR), is reviewed. This section of the SAR (PSAR and FSAR) should describe the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant. Specific information to be reviewed is as follows:

A. An organization chart showing the title of each position, the minimum number of persons to be assigned to common or duplicated positions, the number of operating shift crews, and the positions for which reactor operator and senior reactor operator licenses are required. For multi-unit stations, the organization chart (or additional charts) should clearly reflect changes and additions as new units are added to the station.

B. The schedule, relative to fuel loading for each unit, for filling all positions should be presented.

C. The functions, responsibilities, and authorities of plant positions corresponding to the following should be described:

1. Overall plant management
2. Operations supervision
3. Operating shift crew supervision
4. Licensed operators
5. Non-licensed operators
6. Technical supervision
7. Radiation protection supervision
8. Instrumentation and controls maintenance supervision
9. Equipment maintenance supervision
10. Fire protection supervision

For each position, where applicable, required interfaces with offsite personnel or positions identified in SAR Section 13.1.1 should be described. Such interfaces

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include defined lines of reporting responsibilities, e.g., from the plant manager to his immediate superior, as well as functional or communication channels. In the Final Safety Analysis Report (FSAR), the following should also be described: (1) the line of succession of authority and responsibility for overall station operation in the event of unexpected contingencies of a temporary nature, and (2) the delegation of authority that may be granted to operating supervisors and to shift supervisors, including the authority to issue standing or special orders.

If the station contains or is planned to contain power generating facilities other than those relating to the application in question and including fossil-fueled units, this section should also describe interfaces with the organizations operating such other facilities. The description should include any proposed sharing of persons between the units, a description of their duties, and the proportion of their time they will routinely be assigned to the other unit.

D. The position titles, applicable operator licensing requirements for each, and the total number of people planned to man each shift should be described for all combinations of units proposed to be at the station in either operating or cold shutdown modes Shift crew staffing plans unique to refueling operations should be described. The proposed means of assigning shift responsibility for implementing the radiation protection and fire protection programs on a round-the-clock basis should also be described.

RAB reviews the function, responsibility, authority, and reporting line for the radiation protection supervision against acceptance criteria stated in SRP Section 12.5; the results of this evaluation are reported in Section 12 of the SER.

II. ACCEPTANCE CRITERIA

This section of the SAR should demonstrate the applicant's commitment to (PSAR) and implementation of (FSAR) plans to staff the onsite operating organization and to define and delegate responsibilities to provide assurance that the plant can be operated safely. It should be significant contributory evidence as to the technical qualifications of the applicant, as required by 10 CFR 50.34(a)(9) and (b)(7).

Criteria for acceptability include the following: Note: references to regulatory guides and standards reflect appropriate issue.

A. The requirements of ANSI N18.7, Section 3.4, "Operating Organization," as endorsed by Regulatory Guide 1.33, should be met.

B. Responsibilities and authorities of operating organization personnel should conform to the requirements of ANSI N18.7, Section 5.2, "Rules of Practice"; Section 4.4, "Onsite Review," as endorsed by Regulatory Guide 1.33; Branch Technical Position ASB 9.5-1; ANSI N18.1; Section 3.2, "Operating Organization," as endorsed by Regulatory Guide 1.8; and Regulatory Guide 8.8, Section C.1.b(2) and (3).

C. 1. Assignments of onsite shift operating crews should be made in numbers not less than the following:

For a station having one licensed unit, each shift crew should have at least three persons at all times, plus two additional persons when the unit is operating. For a multi-unit station, each shift crew should have at least three persons per licensed unit at all times, plus one additional person per operating unit.

2. Operator license qualifications of persons assigned to operating shift crews should be as follows:

(a) A licensed senior operator who is also a member of the station supervisory staff should be onsite at all times when at least one unit is loaded with fuel.

(b) For any station with more than one reactor containing fuel, (1) the number of licensed senior operators onsite at all times should not be less than the number of control rooms from which the fueled units are monitored, and (2) the number of licensed senior operators should not be less than the number of reactors operating.

(c) For each reactor containing fuel, there should be at least one licensed operator in the control room at all times. Shift crew compositions should be specified such that this condition can be satisfied independently of licensed senior operators assigned to shift crews to meet the criteria of (a) and (b) above.

(d) For each control room from which one or more reactors are in operation, an additional operator should be onsite and available to serve as relief operator for that control room. Shift crew compositions should be specified such that this condition can be satisfied independently of (a), (b), and (c), and for each such control room.

3. Radiation protection qualifications of at least one person on each operating shift should be as follows:

The management of each station having one or more units containing fuel should either, (1) qualify and designate at least one member of each shift operating crew to implement radiation protection procedures, including routine or special radiation surveys using portable radiation detectors, use of protective barriers and signs, use of protective clothing and breathing apparatus, performance of contamination surveys, checks on radiation monitors, and limits of exposure rates and accumulated dose, or (2) assign a health physics technician to each shift, such assignment to be in addition to those assigned to shift operating crews in accordance with (1) and (2) above.

4. Shift crew assignments for all core alterations after initial fuel loading should include a Licensed Senior Reactor Operator or a Senior Reactor Operator Limited to Fuel Handling with no other concurrent operational duties to directly supervise the core alterations.

5. Assignments of persons to implement the fire brigade requirements of the fire protection program should meet the following:

   (a) The responsibilities of the fire brigade members under normal conditions should not conflict with their responsibilities during a fire emergency.

   (b) The minimum number of fire brigade members available onsite for each shift operation crew should be consistent with the activities required to combat the most significant fire. The minimum size of the fire brigade shift should be five persons unless a specific site evaluation has been completed and some other number justified.

III. REVIEW PROCEDURES

Each element of the SAR information is to be reviewed against this SRP section. The reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the uniqueness of the facility. Any exceptions or alternatives are to be carefully reviewed to assure that they are clearly defined and that an adequate basis exists for acceptance.

The applicant should identify the revision of references, Regulatory Guides, and Codes and Standards used. The reviewer should identify the revision of references, Regulatory Guides, and Codes and Standards used in the review.

In the review and evaluation of the subject matter of this section of the SAR, the following points should be taken into consideration.

Plant staff organizational structures are not rigidly fixed; however, experience has shown that certain components are common to and necessary for all plants. Among these are operational, onsite technical support, and maintenance groups, under the direction and supervision of a plant manager. For multi-unit sites, consideration must be given to the possibility that off-shift supervision may be stretched too thin to provide effective supervision. For example, a single operations manager may have difficulty covering more than two units. For on-shift persons, the total manpower available should be reviewed to assure that in excess of four full operating shift crews are planned so that excessive overtime is not routinely scheduled for these crews. For multi-unit sites, overall site responsibilities should be checked for clarity during those periods of time when senior level supervision is not onsite.

The operating organization, as demonstrated by organization charts and descriptions of functions and responsibilities, should be free of ambiguous assignments of primary responsibility. Operating responsibilities should be reasonably well defined in both numbers and experience of persons required to implement their responsibilities. The

reviewer must recognize that there are many acceptable ways to define and delegate job responsibilities. Variations in staffing may also be expected between applicants who lack prior experience with nuclear plant operation and those who have such experience. It is important that the reviewer assure himself that applicants in the former category do not underestimate the magnitude of the task. The reviewer should be alert to the possibility that excessive workloads may be placed upon too small a number of individuals.

The structure of onsite technical support and maintenance groups may depend somewhat on headquarters staffing and the division of effort between onsite and offsite personnel.

With respect to shift assignments, the reviewer should determine that persons assigned to implement the radiation protection program are adequately trained and qualified for this task, and that it is a clearly defined part of the job function. Assignments to shift crews for refueling operations should be examined to assure adequate supervisory attention is given to all operations associated with fuel handling.

The review procedure for this SRP section consists, therefore, of:

1. An examination of the information submitted to determine that all subject matter identified in Subsection I above has been addressed.

2. A comparison of the information with the acceptance criteria of Subsection II above in the light of the additional points set forth earlier in this SRP section.

3. Review of information provided by the Office of Inspection and Enforcement position statement during the operating license review on the applicant's organizational and administrative commitments.

In addition, if the applicant, as of the time the review takes place, has had experience in the operation of previously licensed nuclear power plants, the reviewer may seek independent information relative to plant staffing and qualifications through the Office of Inspection and Enforcement, e.g., by discussion with inspection personnel or review of inspection reports.

The reviewer then determines, based upon the foregoing, the overall acceptability of the applicant's operating organizations and plant staffing plans.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and its review support conclusions of the following type to be used in the staff's Safety Evaluation Report:

> "The applicant has described the assignment of plant operating responsibilities; the reporting chain up through the chief executive office of the company (applicant); the proposed size of the regular plant staff; the functions and responsibilities of each major plant staff group; and the proposed shift crew complement for single unit

or multiple unit operation. This information has been reviewed, and it is the conclusion of the staff that the proposed organization is acceptable."

V. REFERENCES

A. "Utility Staffing and Training of Nuclear Power," WASH-1130, Revised, U.S. Atomic Energy Commission, June 1973.

B. Regulatory Guide 1.8, "Personnel Selection and Training."

C. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

D. 10 CFR 50.34(a)(6) and (b)(6).

E. U.S. NRC Branch Technical Position ASB 9.5-1.

F. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as is Reasonably Achievable."

VI. APPLICABILITY

The revisions shown in this section of the SRP will be implemented for all new license applications submitted after May 1, 1979.

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

SECTION 13.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

The qualifications established for the applicant's plant personnel as described in its Safety Analysis Report (SAR) are reviewed. This section of the SAR should describe the education, training, and experience requirements established by the applicant for filling each management, operating, technical, and maintenance position category in the operating organization described in SAR Section 13.1.2. At the PSAR stage, it is recognized that many details of the plant organization and staffing have not been finalized. Consequently, the information to be reviewed should demonstrate an understanding of and commitment to the acceptance criteria below. At the Final Safety Analysis Report (FSAR) stage, this section should in addition provide evidence, in the form of personnel resumes, that the initial selections made for management and principal supervisory positions down through the shift supervisory level conform to those requirements.

II. ACCEPTANCE CRITERIA

Regulatory Guide 1.8, "Personnel Selection and Training," sets forth the staff position on plant personnel qualifications and indicates that the criteria for selection (qualifications) contained in ANSI N18.1 are generally acceptable, except as noted in the regulatory position section of Regulatory Guide 1.8. In addition, although the qualification levels of the standard are endorsed as acceptable minimums for each position, it is expected that the collective qualifications of the plant staff will be greater than the sum of the minimum individual requirements described in the standard, particularly in the area of nuclear power plant experience and in supervisory and management positions involved in the operational aspects of the facility. In those cases where the collective qualifications do not exceed the sum of the minimums for individual positions, additional technical support for the plant staff may be required. These will be determined on a case-by-case basis.

III. REVIEW PROCEDURES

Each element of the SAR information is to be reviewed against this SRP section. The reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the magnitude

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and uniqueness of the facility. Any exceptions or alternatives are to be carefully reviewed to assure that they are clearly defined and that an adequate basis exists for acceptance. The applicant should identify the revision of references, regulatory guides, and codes and standards used. The reviewer should identify the revision of references, regulatory guides, and codes and standards used in the review.

At the construction permit (CP) stage, the applicant will generally not have made selections for plant staff positions. The review procedure, therefore, is to examine this section of the SAR for a commitment on the part of the applicant to conform to the stated acceptance criteria. This commitment should be unambiguous and, when such is the case, the review is completed.

Where a clear comparison cannot be made between the proposed plant staff positions and those defined in Section 4 of ANSI N18.1, the applicant should list each position on its plant staff and designate the corresponding position of Section 4 of ANSI N18.1, or describe in detail its proposed qualifications for each position on its plant staff.

The review of the FSAR, at the operating license (OL) stage, consists first of the same examination as made at the CP stage, and secondly, of an analysis of each resume. The reviewer should make an explicit comparison of the educational and experience records obtained from each resume with the corresponding requirements set forth for the applicable position in Section 4 of ANSI N18.1 or other approved qualifications. "Applicable experience" should be judged in the light of the position responsibility. Credit for experience, which may not be entirely applicable, should be weighed to a degree commensurate with its applicability. The bases for such weighed judgments should be documented in the reviewer's notes. When the resumes for persons initially selected to fill all management and principal supervisory positions from the plant superintendent down through each shift supervisor, including technical support personnel, have been analyzed and these persons have been found to have qualifications equal to or greater than those specified in Regulatory Guide 1.8 and other approved qualifications, the review is completed.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and its review support conclusions of the following type to be used in the staff's safety evaluation report:

1. PSAR

   "The applicant has described the qualification requirements for members of its plant staff. This information has been reviewed, and we conclude that the qualifications conform to the staff positions set forth in Regulatory Guide 1.8."

2. FSAR

   "The applicant has described the qualification requirements for members of its plant staff and submitted personnel resumes for management and principal supervisory positions.

"This information has been reviewed, and we conclude that the qualifications conform to staff positions set forth in Regulatory Guide 1.8 and that the selection of initial appointees to the plant staff conforms to these requirements and is equal to or greater than the qualification requirements."

V. REFERENCES

Regulatory Guide 1.8, "Personnel Selection and Training."

VI. APPLICABLITY

The revisions shown in this section of the SRP will be implemented for all new license applications submitted after May 1, 1979.

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 13.2 TRAINING

REVIEW RESPONSIBILITIES

Primary - Operator Licensing Branch (OLB)

Secondary - None

I. AREAS OF REVIEW

The applicant's plant personnel training program, as described in his Safety Analysis Report (SAR), is reviewed. This section of the SAR should contain the description and scheduling of the training program for initial appointees to the plant staff. The program descriptions should include the following:

For the Preliminary Safety Analysis Report (PSAR):

1. The proposed subject matter of each course, the duration of the course (approximate number of weeks in full time attendance), the organization teaching the course or supervising instruction, and the position titles for whom the course is given.

2. Reactor operations experience training by nuclear power plant simulator or assignment to a similar plant, including length of time (weeks), and identity of simulator and plant.

3. A commitment to conduct an onsite formal training program and on-the-job training before the initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to Section 55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following:

   a. Individuals with no previous experience.

   b. Individuals who have had nuclear experience at facilities not subject to licensing.

   c. Individuals who hold, or have held, licenses for comparable facilities.

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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. A commitment to conduct an initial fire protection training program, including:

   a. Periodic drills during construction

   b. Provisions for indoctrination of construction personnel, as necessary.

   The initial training shall be completed prior to receipt of fuel at the site.

6. A detailed description of the training program for the individual(s) responsible for the formulation and assurance of the implementation of the fire protection program. The training program should address those items listed in Section A.1 of Appendix A to Branch Technical Position ASB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants.

7. The means for evaluating the training program effectiveness for all employees. For license applicants this includes the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to Section 55.25(b) of 10 CFR Part 55.

   This program description should also include a chart to show the schedule of each part of the training program for each functional position identified in SAR Section 13.1.2. The time should be relative to expected fuel loading and should also display the preoperational test period, and the expected time for examinations for licensed operators prior to plant criticality.

In the Final Safety Analysis Report (FSAR):

1. The proposed subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks in full time attendance), the organization teaching the course or supervising instruction, and the position titles for which the course is given.

2. Reactor operations experience training by nuclear power plant simulator or assignment to a similar plant, including length of time (weeks), and identity of simulator and plant.

3. The details of the onsite training program, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks in full time attendance), the organization teaching the course or supervising instruction, and the position titles for which the course is given. The program should distinguish between classroom training and on-the-job training, before and after the initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to Section 55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following:

   a. Individuals with no previous experience.

   b. Individuals who have had nuclear experience at facilities not subject to licensing.

   c. Individuals who hold, or have held, licenses for comparable facilities.

5. A detailed description of the fire protection training and retraining for the initial plant staff and replacement personnel. The program shall describe:

   a. The training planned for each member of the fire brigade,

   b. The type and frequency of periodic firefighting drills,

   c. The training provided for all remaining staff members, including personnel responsible for maintenance and inspection of fire protection equipment,

   d. The indoctrination and training provided for persons temporarily assigned onsite duties during shutdown and maintenance outages, particularly those allowed unescorted access,

   e. The training provided for the fire protection staff members.

   The description shall include the courses of instruction and number of hours of each course and the organization conducting the training.

6. Means for evaluating the training program effectiveness for each employee. For applicants for license examinations prior to criticality, the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to Section 55.25(b) of 10 CFR Part 55.

The program description section should also include a chart to show the schedule of each part of the training program for each functional position identified in FSAR Section 13.1.2. The time scale should be relative to expected fuel loading and should also display the preoperational test period, expected time for examinations for licensed operators prior to criticality, and expected time for examinations for licensed operators after criticality (13.2.1.2).

The description should delineate clearly the extent to which the training program has been accomplished at the approximate time of submittal of the FSAR. Contingency plans for additional training for individuals to be licensed prior to criticality should be described in the event fuel loading is subsequently delayed from the date indicated in the FSAR.

The FSAR should describe the applicant's plans for retraining of plant staff personnel including requalification training for licensed operators and senior operators (13.2.2). The detailed description of the proposed requalification training program should show how it will meet the requirements of 10 CFR Part 55, Appendix A (13.2.2.1). The FSAR should also identify the additional position categories on the plant staff for which retraining will be provided, and should describe the nature, scope, and frequency of such retraining (13.2.2.2).

II. ACCEPTANCE CRITERIA

The SAR should demonstrate that the training provided, or to be provided, for each position on the plant staff will be adequate to provide assurance that all plant staff personnel qualification requirements will be met as of the time needed, i.e., prior to operator license examinations, prior to fuel loading, or prior to appointment or reappointment to the position.

Criteria for acceptability are:

1. The training requirements and guidance set forth in the following regulations and regulatory guides should be met or acceptable alternatives should be presented.

10 CFR Part 50	All Employees
10 CFR Part 19	All Employees
Regulatory Guide 1.8	All Employees
Regulatory Guide 1.101	All Employees
Branch Technical Position ASB 9.5-1	All Employees
10 CFR Part 55	Licensed Operators and Senior Operators
NRC Operator Licensing Guide, NUREG-0094	Licensed Operators and Senior Operators

2. Formal segments of the initial training program should be substantially completed when the preoperational test program begins, with the exception of a brief, formal refresher just prior to operator examinations.

3. The number of persons for whom training is planned in preparation for senior operator and operator examinations prior to criticality should be sufficient to assure that applicable technical specification conditions with respect to the

number of licensed operators on shift crews can be met from the time of initial fuel loading of the first unit, with due allowance given for examination contingencies and the need to avoid planned overtime for supervisory personnel during the startup phase in order to meet technical specification conditions.

4. The licensed operator requalification training program should adequately implement the requirements of 10 CFR Part 55, Appendix A.

5. Refresher training for non-licensed personnel should be periodic and not less than biennial and should include at a minimum, refresher instruction on administrative, radiation protection, emergency, and security procedures.

6. Fire Protection Training

A. Fire Brigade Training

1. Instruction

a. Instruction in all the topics listed in d below should be administered to individuals prior to assignment as a fire brigade member.

b. Refresher instruction should be provided to all fire brigade members on a regularly scheduled basis of not less than four sessions a year. The sessions shall be repeated at a frequency of not more than 2 years.

c. The instruction shall be provided by qualified individuals, knowledgeable and experienced in fighting the types of fires that could occur in the plant and in using the types of equipment available in a nuclear power plant. Members of the Fire Protection Staff and fire brigade leaders may conduct this training.

d. The scone of the instruction should include the following items.

(i) An identification of the fire hazards and associated types of fires that could occur in the plant, and an identification of the location of the hazards, including areas where breathing apparatus is required, regardless of the size of the fire.

(ii) Identification of the location of installed and portable fire fighting equipment in each area, and familiarization with layout of the plant including access and egress routes to each area.

(iii) The proper use of available equipment, and the correct method of fighting each type of fire. The types of fires covered should include electrical fires, fires in cables and cable trays, hydrogen fires, flammable liquids, waste/debris fires, and record file fires.

(iv) Indoctrination in the plant fire fighting plan, with coverage of each individual's responsibilities, including changes thereto.

(v) The proper use of breathing equipment, communication, lighting and portable ventilation equipment.

(vi) A detailed review of the procedures, with particular emphasis on what equipment must be used in particular areas.

(vii) A review of latest modifications, additions or changes to the facility, procedures, fire fighting equipment or fire fighting plan.

(viii) The proper method of fighting fires inside buildings and tunnels.

e. In addition, special instruction should be provided for fire brigade leaders in directing and coordinating fire fighting activities.

2. Practice

Practice sessions should be held for fire brigade members on the proper method of fighting various types of fires. These sessions should provide brigade members with practice in extinguishing actual fires, except in the case of energized cables. Practice sessions should be conducted at facilities sufficiently remote from the nuclear power plant so as not to endanger safety-related equipment. These practice sessions should be provided at regular intervals, but not to exceed one (1) year.

Practice sessions should also be conducted that require fire brigade members to don protective equipment, including emergency breathing apparatus. These practice sessions need not include fire fighting. These practice sessions should be provided at regular intervals, but not to exceed one (1) year.

3. Drills

Fire brigade drills should be performed in the plant so that a fire brigade can practice as a team. Drills should include the following.

a. The simulated use of equipment for the various situations and types of fires which could reasonably occur in each safety-related area.

b. Conformance, where possible, to the established plant fire fighting plans.

c. Operating fire fighting equipment where practical. This would also include self-contained breathing apparatus, communication equipment and portable and/or installed ventilation equipment.

d. The drills should be performed at regular intervals, but not to exceed three months for each fire brigade. The minimum number of fire brigade drills conducted within a period of three months shall be equal to the number of operating shifts at the station. Each individual member of the fire brigades shall participate in at least two drills per year. At least one drill per year for each fire brigade shall be unannounced.

e. Periodically (at least annually), these drills should include off-site fire department personnel. These drills should also conform with the facility plan for coordination with off-site fire departments.

f. The drills should be preplanned to establish the training objectives of the drills. The drills should be critiqued to determine how well the training objectives have been met. At a minimum, the critique should assess:

(i) Fire alarm effectiveness, response time, selection, placement and use of equipment.

(ii) The leader's direction of the effort and each member's response.

B. Other Station Employees

1. Instruction

a. Instruction shall be provided for all employees once a year. It shall be repeated on an annual basis. The instruction shall be given, as appropriate, on (a) the fire protection plan (b) evacuation routes and (c) procedure for reporting a fire.

b. Instruction shall be provided for security personnel that addresses (a) entry procedures for outside fire departments (b) crowd

control for people exiting the station, and (c)procedures for reporting potential fire hazards observed when touring the facility.

c. Instruction should be provided to all shift personnel that complements that provided members of the fire brigade.

d. Instruction shall be provided to temporary employees so that they are familiar with (a) evacuation signals, (b) evacuation routes and (c) procedure for reporting fires.

2. Drills

All employees should participate in an annual evacuation drill.

C. Fire Protection Staf:

Training for the fire protection staff members include courses in:

1. design and maintenance of fire detection, suppression and extinguishing systems,

2. fire prevention techniques and procedures,

3. training and manual firefighting techniques and procedures for plant personnel and the fire brigade.

D. Off-Site Fire Departments

Training for the off-site fire departments include courses in basic radiation principles and practices, typical radiation hazards that may be encountered when fighting fires and related procedures.

E. Construction Personnel

Training for construction personnel should include instruction in reporting fires, alarm responses and evacuation routes.

III. REVIEW PROCEDURES

Preparation for the review of Section 13.2 of the SAR should include familiarization with 10 CFR Part 50, 10 CFR Part 55, 10 CFR Part 19, Regulatory Guide 1.8, and NUREG-0094, "NRC Operator Licensing Guide," Regulatory Guide 1.101 and Branch Technical Position ASB 9.5-1. The reviewer may use training course descriptions obtained independently from vendors.

The review procedure for this SRP Section consists of:

1. A careful examination of the information submitted to determine that all subject matter identified in Subsection I above has been addressed, and

2. A detailed comparison of the information with the acceptance criteria of Subsection II above.

The reviewer should ensure that whenever the applicant has committed to follow the position of a regulatory guide, industry standard or other reference document, the specific revision being referred to is identified. Similarly, whenever the reviewer is using a position in a reference document as a basis for acceptability, the revision being made should be identified.

The reviewer then determines, based upon the foregoing, the overall acceptability of the applicant's plant staff training plans.

IV. EVALUATION FINDINGS

The reviewer should verify that the information presented and his review support an evaluation findings statement of the following type, to be used in the staff's safety evaluation report:

"The training program and schedule for all staff members is acceptable for the preoperational test program, for operator licensing examinations, and for fuel loading."

"Plans for retraining and replacement training conform to the (named) regulatory position, or equivalent, and the commitment to replacement training will be incorporated in the Administrative Controls section of the applicant's Technical Specifications."

"Plans for fire protection training and retraining of the fire protection staff, the fire team (fire brigade) personnel and other staff personnel and plans for periodic drills provide reasonable assurance that the plant staff will be adequately trained to cope with fire-related emergencies. Construction workers and temporarily assigned personnel will be sufficiently indoctrinated so that they contribute to the overall fire protection program."

The evaluation findings should also include the following information:

1. Identity of organization(s) that will conduct or have conducted designated parts of the initial training program.

2. Identity of major training subject areas.

V. REFERENCES

1. 10 CFR Part 50, "Licensing of Production and Utilization Facilities."

2. 10 CFR Part 55, "Operators' Licenses."

3. 10 CFR Part 19, "Notices, Instructions, and Reports to Workers; Inspections."

4. Regulatory Guide 1.8, "Personnel Selection Training."

5. "NRC Operator Licensing Guide," NUREG-0094, July 1976.

6. "Utility Staffing and Training for Nuclear Power," WASH-1130, USAEC, Revised June 1973.

7. Regulatory Guide 1.101, "Emergency Planning."

8. Branch Technical Position ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants."

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SECTION 13.3 EMERGENCY PLANNING

REVIEW RESPONSIBILITIES

Primary - Emergency Planning Branch (EPB)

Secondary - None

I. AREAS OF REVIEW

The applicant's emergency planning, as described in his safety analysis report (SAR), is reviewed by EPB. This primary review responsibility involves evaluation of evidence of preliminary planning (in the preliminary safety analysis report, PSAR) or substantive evidence of planning (in the final safety analysis report, FSAR) for emergency preparedness directed primarily at situations involving real or potential radiological hazards. This review covers each of the sub-parts of 10 CFR Part 50, Appendix E, applicable to the particular licensing stage, and the information requested in Section 13.3 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

For applications made pursuant to 10 CFR Part 50, Appendices M and O, EPB reviews those elements of facility design features that affect plans for coping with emergencies that meet the criteria identified below in II.A.3.(O)(1) and II.A.3(G)(1)-(3), to the extent that such features fall within the scope of the application.

II. ACCEPTANCE CRITERIA

A. PSAR Stage

The reviewer should conclude that an applicant's preliminary plans for coping with emergencies are acceptable if:

1. They are compatible with facility design features, site layout, and site location,

2. They provide reasonable assurance that appropriate protective measures can be taken in the event of an emergency, within and outside the site boundary, and

3. They conform to the requirements of 10 CFR Part 50, Appendix E (II), and to the following supplementary criteria which are identified to clarify certain details of the review. [The alphabetic sequence corresponds to the sub-paragraphs of Appendix E (II).]

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.

(A) Emergency Organization

(1) Plant staff emergency organization assignments and headquarters support assignments should be compatible with and reasonably correlated to the managerial and functional responsibilities that are applicable to non-emergency assignments.

(2) Interfaces between and among the various organizational components should be identified and defined, e.g., by means of a block diagram.

(3) Proposed arrangements for notifications to persons assigned to the emergency organization should include both procedural matters and hardware systems.

(B) Contacts and Arrangements

(1) Arrangements with the principal State agency should reflect the adoption, or tentative adoption of (a) emergency notification criteria, and (b) protective action criteria (Protective Action Guides) for the public outside the site boundary.

(2) The principal government office or agency in each local political jurisdiction (county and municipality) within the low population zone, which would have the responsibility for prompt implementation of protective action warnings and instructions to the public, should be clearly identified. Evidence that preliminary arrangements have been undertaken by the applicant with each such agency, either directly or through a cognizant State agency, should be included.

(C) Emergency Measures

(1) Preliminary planning should reflect the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including an in-plant capability for dose projection for plume exposure pathways (cf. Reg. Guide 1.97). The planning should reflect, in particular, commitments to the following:

(a) For an accident of the type postulated pursuant to 10 CFR Part 100, the initial assessment measures should assure a capability for prompt notification to offsite authorities, i.e., within approximately fifteen minutes of detection of the initiating event.

(b) Real-time meteorological information obtained at the site and a specified source of timely weather forecasting information should be incorporated in accident assessment procedures.

(c) The plans should provide for the timely dispatch of radiological monitoring teams to assess direct radiation exposure pathways from plant and ground deposition sources within the site boundary and outside the site boundary at least to the low population zone boundary.

(d) The plans should provide for taking precipitation, water, and vegetation samples within the site boundary and outside at least to the low population zone boundary to assess potential ingestion exposure pathways.

(2) The expected response and anticipated capabilities of offsite agencies for radiological hazards assessment in the environs should be described, with particular emphasis on ingestion exposure pathways.

(3) Preliminary planning should reflect the need to include provisions for corrective actions at the plant in the event of an accident, including shutting down equipment, fire fighting, and repair and damage control, under potentially adverse conditions.

(4) Preliminary planning should reflect provisions for initiating protective actions for all exposure pathways, onsite and offsite, including:

(a) Direct radiation exposure from a confined source in-plant, an airborne plume, and ground deposition,

(b) Inhalation exposure from an airborne plume, and

(c) Ingestion exposure from contaminated water, milk, and other agricultural products.

(5) Plans for protective actions within the site boundary should include evacuation of all persons not having emergency duty assignments, sheltering, respiratory protection, protective clothing, contamination control, and area access control.

(6) Plans for protective actions outside the site boundary should include evacuation and area access control. The region within which these provisions should be made normally encompass the low population zone. If supported by an assessment of the specific plant and site

situation, this region should be extended beyond the LPZ. The principal criteria to be employed in this regard include (a) a determination that a relatively large number of transient persons may be present within approximately five miles of the plant for whom there is little or no opportunity for sheltering, and (b) a comparison of projected doses from postulated releases with Federal guidance on those protective action criteria which might mandate the initiation of an evacuation, i.e., upper limit values where ranges are given. Calculated doses for the early hours (up to about eight) following onset of a postulated release should be considered when applying this latter criterion.

(D) First Aid and Decontamination Facilities; Emergency Transportation

(1) Proposed locations and physical layouts of first aid and personnel decontamination facilities in the plant should be indicated. Brief descriptions of proposed equipment and supplies for these facilities should be included.

(2) Proposed arrangements for transportation of injured personnel from the site to a relatively nearby hospital should be adequate to accommodate at least two persons who may also be radiologically contaminated, within approximately one hour.

(E) Hospitals

(1) The identification of offsite hospital facilities should include:

(a) One facility relatively nearby which is willing and potentially capable of handling patients from the plant site who may be radiologically contaminated, on an emergency basis, and

(b) A second hospital which is expected to be capable of providing definitive patient care for acute radiation injury.

(F) Training

(1) The scope and content of training to be provided to each member of the emergency organization should be tailored to the assigned emergency duties with emphasis on those matters which each individual is not likely to experience in his normal occupational function.

(2) Each offsite organization or individual proposed to provide special support services, i.e., ambulance, medical, hospital, and fire department, should be identified in the proposed training program.

(G) Plant Evacuation and Reentry

(1) Communications systems needed for plant evacuation and reentry should include an evacuation alarm, a fire alarm, and voice communications systems.

(2) The tentative identification and proposed utilization of one or more emergency operations centers should be provided.

(3) Assurance should be given that features for plant evacuation and reentry will be compatible with the plant security system and procedures.

B. FSAR Stage

The reviewer should conclude that an applicant's plans for coping with emergencies are acceptable if:

1. The plans conform to the requirements of 10 CFR 50, Appendix E (IV),

2. They meet the Regulatory Position Statement in Section C of Regulatory Guide 1.101, (applicable revision),

3. They provide an adequate basis for the Commission's inspection and enforcement program relative to emergency preparedness throughout the lifetime of the facility, and

4. They provide reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect public health and safety and prevent damage to property. [10 CFR 50, Appendix E (III)].

III. REVIEW PROCEDURES

Following the acceptance of each SAR, the review is conducted on a schedule which is established for each SAR. It consists of an evaluation of the emergency planning information submitted by the applicant utilizing the foregoing Acceptance Criteria. Although the bulk of this information should be found in Section 13.3 (or referenced therein*) the reviewer should gain familiarity with the site, including the exclusion area, low population zone, population center, demography, and land use factors, with the proposed plant design and layout, and with the calculated dose consequences of design basis accidents postulated by the applicant. To this end, the reviewer should examine relevant sections of the SAR, particularly sections found in Chapters 1, 2, 6, 7, 9, 11, and 15. The reviewer should also gain familiarity with proposed radiation protection activities and other operational matters that interface with emergency plans, particularly as described in the SAR in sections of Chapters 12 and 13. Draft and final environmental

*Including, as applicable, reference SSAR's for which a PDA or FDA has been issued.

statements for the proposed facility should also be consulted when available during the review process. This information may be supplemented by a personal visit to the site by the reviewer and meetings with the applicant. In cases where the applicant is a licensee for a previously licensed plant, NRC inspection reports should be reviewed. For each case, formal consultation with NRC's Office of State Programs with respect to the relevant State and local government emergency response capabilities is necessary.

For each case assigned to him, the reviewer is expected to determine whether or not the acceptance criteria identified in II above have been satisfactorily met. Deviations from these criteria should form the basis for requests for additional information or transmittal of position statements to the applicant, and should be reviewed with the Branch Chief. Such further review may result in a determination that (a) the applicant has proposed acceptable alternatives, (b) the facts of the case do not warrant the application of the criterion in question, or (c) the facts do warrant the application of criterion in question and no acceptable alternative has been proposed or identified. If any such deviations remain in the last category at the conclusion of the review, they must be identified in the Safety Evaluation Report and subsequently resolved with the participation of higher level NRC management.

It should be recognized that the detailed application of the acceptance criteria will in many instances require the exercise of judgment on the part of the reviewer. The reasonableness and adequacy of the factors involved should be viewed in the light of general emergency planning and response experience, bearing in mind that the broad objective of radiological emergency plans is to benefit and protect the public by mitigating the potential health and safety consequences of radiation exposure in a rational manner. Ideally, such plans would assure neither an over reaction nor an under reaction to unexpected events. Reviewers should be particularly alert, however, to provisions which may result in a possible under reaction to a serious event.

In addition to the foregoing, at the PSAR stage, the reviewer should assess the applicant's analysis and findings relative to emergency planning for evacuation. The reviewer should consult with the Accident Analysis Branch relative to the applicant's identification and treatment of the most serious design basis accident. Particular attention should be paid to the credibility and adequacy of the time estimates or allocations for sequential actions. Independent estimates of evacuation times should be obtained.

At each licensing stage the reviewer should coordinate his review of (1) exclusion area and low population zone suitability matters with the Accident Analysis Branch, and (2) accident assessment instrumentation and methods with other Branches having responsibility for evaluations relative to Regulatory Guide 1.97. At the FSAR stage the reviewer should coordinate his review of fire emergency plans with the Auxiliary Systems Branch, and emergency communications matters with the Power Systems Branch, and should consult with the Reactor Safeguards Licensing Branch to assure compatibility between emergency plans and safeguards plans.

At the beginning of the FSAR stage review the reviewer should examine the Construction Permit docket record, including PSAR, staff Safety Evaluation Report(s), recommendations of the Advisory Committee on Reactor Safeguards, and the public hearing record, for information that may bear on the FSAR review of plans for coping with emergencies. At the FSAR stage for multi-unit sites the reviewer should carefully distinguish whether the plans are applicable only to the first unit or to subsequent units as well.

At the conclusion of the review, findings on acceptability of the applicant's proposed exclusion areas and low population zone, with respect to emergency planning considerations, should be transmitted to the Accident Analysis Branch. Findings on the acceptability of proposed plans for coping with emergencies should be prepared for input to the staff's Safety Evaluation Report.

IV. EVALUATION FINDINGS

The desired evaluation findings at the PSAR stage should be substantially equivalent to the following statement:

"We have evaluated the applicant's preliminary plans for coping with emergencies as presented in Chapter 13.3 of the PSAR through revision/amendment ___. (As necessary add specific references to docketed material reviewed as a result of staff questions and/or positions.) We find that the preliminary plans are acceptable in that they meet or exceed the minimum requirements of 10 CFR Part 50, Appendix E, conform to the staff's acceptance criteria, are consistent with facility design features, analyses of postulated accidents, and characteristics of the proposed site location, and provide reasonable assurance that appropriate protective measures can be taken within and outside the site boundary in the event of an emergency."

(The following paragraphs should summarize the specific bases for the finding, including how the plans meet each of the elements A through G of 10 CFR Part 50, Appendix E, sub-paragraph II, and the results of the consultation with the NRC Office of State Programs.)

"The staff will evaluate the applicant's final plans for coping with emergencies at the FSAR stage, to be submitted in accordance with 10 CFR Part 50.34(b)."

The desired evaluation finding at the FSAR stage should be substantially equivalent to the following:

"We have evaluated the applicant's plans for coping with emergencies as presented in Chapter 13.3 of the FSAR through revision/amendment____. These plans are documented in (specific references to the docketed material reviewed including, as necessary, those submitted as a result of staff questions and/or positions.) We find that these plans are acceptable in that they meet or exceed the minimum requirements of 10 CFR Part 50, Appendix E, conform to the staff's acceptance criteria as set forth in Regulatory Guide 1.101 (Revision __), and that they provide reasonable assurance that appropriate measures

can and will be taken in the event of an emergency to protect public health and safety and prevent damage to property."

(The following paragraphs should summarize the specific bases for the finding including how the plans meet each of the elements A through J of 10 CFR Part 50, Appendix E, sub-paragraph IV, and the results of the consultation with the Office of State Programs as to the emergency response capabilities of the relevant State and local government agencies.)

"The applicant's plans include measures for coping with fire emergencies that conform with the applicable provisions of Regulatory Guide 1.101. In particular, a satisfactory written agreement is in effect with the [identity of offsite fire department(s)] which assures the availability of additional trained personnel and equipment for fire fighting support when called upon. In addition, the applicant has provided for annual training of these personnel to assure their necessary familiarity with the plant, access procedures, and radiation protection precautions, and for their participation in an annual drill or test exercise."

V. REFERENCES

1. Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants."

2. 10 CFR Part 50, Appendix E, "Emergency Plans for Production and Utilization Facilities."

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

4. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident.

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SECTION 13.4 OPERATIONAL REVIEW

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

QAB reviews and evaluates the applicant's plan for conducting reviews of operating phase activities that are important to safety, as described in the applicant's Final Safety Analysis Report (FSAR). The primary focus of attention should be on the provisions that will be used to implement the licensee's responsibility pursuant to 10 CFR 50.59 relating to proposed changes, tests, and experiments, and on the procedures for after-the-fact review evaluation of unplanned events, such as Licensee Event Reports. No information is required in the PSAR.

The FSAR should describe provisions for review by plant staff members and for independent review of plant operations. Specific information to be reviewed is as follows:

1. How the onsite organization functions with respect to review of proposed changes to systems or procedures, tests, and experiments, and of unplanned events that have operational safety significance. This will include subject matter to be reviewed, organizational provisions for conducting the reviews including personnel, and the documentation and reporting of review activities.

2. The procedure and organization employed to examine safety-related operating activities independent of the operating organization. This will include how and when such a program is to be implemented, relative to fuel loading of the first unit, and include subject matter to be reviewed, organizational provisions for conducting the review including personnel, and the documentation and reporting of review activities.

II. ACCEPTANCE CRITERIA

The staff positions applicable to this section are as follows for 1 and 2 above respectively: (Note-Below references to regulatory guides and standards reflect appropriate issues.)

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.

1. Plant Staff Review

   a. Scope of this review should include that of 10 CFR 50.59 and Section 4.4 of ANSI N18.7 as endorsed by Regulatory Guide 1.33.

   b. Organizational arrangements should provide for interdisciplinary reviews of subject matter.

   c. Qualification levels for plant staff personnel performing review should be at least equivalent to those described in Section 4.4 of ANSI N18.1 as endorsed by Regulatory Guide 1.8.

   d. Review activities should be documented and results forwarded to appropriate members of management.

2. Independent Review

   Provisions for independent review should meet that described in Sections 4.1 - 4.3 of ANSI N18.7.

III. REVIEW PROCEDURES

Each element of the FSAR information is to be reviewed against this SRP section. The reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the magnitude and uniqueness of the project. Any exceptions or alternatives are to be carefully reviewed to assure they are clearly defined and that an adequate basis exists for acceptance. The applicant should identify the revision of references, regulatory guides and codes and standards used. The reviewer should identify the revision of references, regulatory guides and codes and standards used in the review.

The review of this section of the FSAR consists of an analysis of the information submitted by detailed comparison with the acceptance criteria of subsection II, above. When the reviewer has determined that the acceptance criteria stated above or their equivalent have been satisfactorily addressed in the applicant's plans for conducting reviews, the review of this section of the SAR is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review support conclusions of the following type, to be used in the staff's safety evaluation report: (Note-The appropriate issue of the regulatory guide and standard should be stated.)

"The applicant's program for the review of plant operations is in conformance with staff positions described in Regulatory Guide 1.33 and applicable industry standard (ANSI N18.7), and is acceptable."

V. REFERENCES

1. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

2. Regulatory Guide 1.8, "Personnel Selection and Training."

3. 10 CFR 50.59, "Authorization of Changes, Tests and Experiments."

VI. APPLICABILITY

Revision 1 of this SRP Section will be applied to all operating license applications docketed after April 30, 1979.

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SECTION 13.5 PLANT PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Operator Licensing Branch (OLB)

Secondary - None

I. AREAS OF REVIEW

OLB reviews the plant procedures, as described in the applicant's safety analysis report (SAR). This section of the SAR should describe administrative and operating procedures that will be used by the operating organization (plant staff) to assure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, it is not expected that detailed written procedures will be included in the SAR. The preliminary safety analysis report (PSAR) should describe preliminary schedules for their preparation and the final safety analysis report (FSAR) should provide descriptions as to the nature and content of procedures as detailed below.

1. The SAR section on administrative procedures (13.5.1) should include a commitment to conduct all safety-related operations by detailed written and approved procedures, and should provide for the preparation of written administrative procedures which will govern the safety-related activities of the plant staff. The FSAR should identify the persons (by title and organization) having responsibility for the writing of procedures, and the persons who must review and approve them before they are implemented. In the FSAR, a description of administrative procedures should be provided, which would include the following:

   a. Standing orders to operations shift supervisors and shift crews including:

      (1) The reactor operator's authority and responsibilities.

      (2) The senior operator's authority and responsibilities.

      (3) The responsibility to meet the requirements of 10 CFR §50.54(i), (j), (k), (l) and (m), including a diagram of the control area that indicates the area designated "at the controls."

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.

b. Special orders of a transient or self-cancelling character.

c. Equipment control procedures.

d. Control of maintenance and modifications.

e. Master surveillance testing schedule.

f. Temporary procedures.

g. Fire Protection Procedures.

2. A section on operating and maintenance procedures should be included in the FSAR (13.5.2).

a. The first part should deal with procedures which are performed by licensed operators in the control room. Each such operating procedure should be identified by title and included in a described classification system. The general format and content for each class should be described. The following categories should be included, but need not necessarily form the basis for classifying these procedures:

(1) System procedures.
(2) General Plant procedures.
(3) Off-normal operating procedures.
(4) Emergency procedures.
(5) Alarm response procedures.
(6) Temporary procedures.

In category (5), individual alarm response procedures should not be listed. However, the system employed to classify or subclassify alarm responses, and the methods to be employed by operators to retrieve or refer to alarm response procedures should be described. Immediate action procedures required to be memorized should be identified.

b. The second part of this section should describe how other operating and maintenance procedures are classified, what group or groups within the operating organization have the responsibility for following each class of procedures, and the general objectives and character of each class and subclass. The following categories should be included. If their general objectives and characters are described elsewhere in the FSAR, or application, these may be described by specific reference thereto.

(1) Plant radiation protection procedures.
(2) Emergency preparedness procedures.

(3) Instrument calibration test procedures.
(4) Chemical and radiochemical control procedures.
(5) Radioactive waste management procedures.
(6) Maintenance and modification procedures.
(7) Materials control procedures.
(8) Plant security procedures.
(9) Fire protection procedures

II. ACCEPTANCE CRITERIA

This section of the SAR should demonstrate the applicant's commitment to the conduct of operations by means of written and approved procedures. It constitutes additional evidence of his technical qualifications, and forms a basis for a key part of the regulatory inspection program.

A generally acceptable target date for completion of administrative procedures and operating procedures is about six months before fuel loading, inasmuch as familiarization with these procedures is an essential part of the staff training program, including preparation for operator license examinations prior to criticality.

This section of the FSAR should comply with the guidance contained in Regulatory Guide 1.33 or present acceptable alternatives. In addition, this section of the FSAR should indicate the plans for meeting the requirements of §50.54(i), (j), (k), (l), and (m) of 10 CFR Part 50.

III. REVIEW PROCEDURES

The review under this SRP section consists of a detailed comparison of the information submitted with the acceptance criteria of Subsection II above, as applicable to the PSAR or FSAR. When the reviewer has determined that each of these criteria has been satisfied, based upon the statements made by the applicant in the SAR, the review under this SRP section is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review, support the following type of conclusion to be used in the staff's safety evaluation report:

"The applicant's provisions for administrative procedures and operating procedures conform to this guidance contained in Regulatory Guide 1.33 and applicable industry standards and are acceptable."

"The applicant's provisions meet the requirements of §50.54(i), (j), (k), (l), and (m) of 10 CFR Part 50."

"The significant provisions will be included in the administrative controls section of the plant's technical specifications."

The evaluation findings for this SRP section should also include the following:

1. A statement that plant operations are to be performed in accordance with written and approved procedures.

2. A brief description of the categories of procedures to be included.

3. A brief description of the review and approval mechanism for procedures and changes, thereto.

V. REFERENCES

1. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

2. 10 CFR Part 50, §50.54, "Conditions of Licenses."

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Section 13.6 INDUSTRIAL SECURITY

REVIEW RESPONSIBILITIES

Primary - Industrial Security and Emergency Planning Branch (ISEPB)

Secondary - None

I. AREAS OF REVIEW

At the preliminary safety analysis report (PSAR) stage, the review of this section covers plans for implementing security measures relating to (1) the screening of personnel employed to work at the proposed plant and (2) the layout of the plant and other design features and equipment arrangements intended to provide protection of vital equipment against acts of industrial sabotage.

At the final safety analysis report (FSAR) stage, the review involves the evaluation of the industrial security plan, which describes a comprehensive physical security program for the plant site. The review encompasses the physical security organization, access controls to the plant including physical barriers and means of detecting unauthorized intrusions, provisions for monitoring the status of vital equipment, selection and training of personnel for security purposes, communications systems for security, and arrangements with law enforcement authorities for assistance in responding to security threats. The implementation schedule for the physical security program is reviewed, including phases for multi-unit plants where applicable.

Specific information to be reviewed, referenced to applicable sections of ANSI N18.17-1973, include the following:

1. Clear diagrams, to approximate scale, displaying the following:
   a. Designated security areas of the plant site, including physical barriers.
   b. The locations of alarm stations.
   c. The locations of access control points to protected and to vital areas.
   d. The location of parking lots relative to the clear areas adjacent to the physical barriers surrounding protected areas.
   e. Special features of the terrain which may present special vulnerability problems.

USNRC STANDARD REVIEW PLAN

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f. The location of relevant law enforcement agencies and their geographical jurisdictions.

2. If the policy of the owner organization permits use of any part of the owner-controlled area by members of the general public, details of how the requirements of Section 3.2 will be met.

3. The response capabilities of local law enforcement agencies (Section 4.4.7), including estimates of the number of officers that can arrive at the plant site in the event of a security threat, within five to fifteen minutes, fifteen to thirty minutes, and thirty minutes to one hour, after receipt of a call for assistance. (This response capability bears upon the adequacy of the size of the onsite guard force.)

II. ACCEPTANCE CRITERIA

At the PSAR stage, preliminary planning for industrial security should show how conformance to the applicable provisions of Regulatory Guide 1.17 are expected to be achieved, including:

1. ANSI N18.17-1973, Section 2, "Definitions;" Section 3, "Designated Security Areas;" Section 4.3, "Employee Screening"; and Section 5, "Plant Design."

2. Regulatory Guide 1.17, Revision 1, Section C.1.b, "Security Alarms," and Section C.3, "Protection of Vital Equipment."

This planning should include a commitment to design phase review for physical security and should show how this responsibility is to be implemented by the applicant.

At the FSAR stage, the applicant's security plan must conform to the requirements of 10 CFR 50.34(c), and to applicable requirements of 10 CFR Part 73. In addition, the provisions of Regulatory Guide 1.17, Revision 1, including the requirements and recommendations of ANSI N18.17-1973, Sections 3 and 4, establish the basis for an adequate security plan for the protection of nuclear power plants against industrial sabotage.

Specific acceptance criteria, including staff interpretations of some of the more general requirements of the ANSI Standard, are as follows: (Section references are to sections of ANSI N18.17-1973.)

1. Surveillance of a protected area (Section 3.3.3) should be by a system which can provide for continuous monitoring of the entire perimeter of a protected area so as to allow response to be initiated at the time of penetration of a protected area.

2. Central alarm stations should be regarded as vital areas and meet the qualifications required thereof.

3. For each plant, the onsite security force should include not less than two guards on each shift.

4. "Armed guards" means guards physically carrying firearms. Persons assigned to control access points to protected areas should not be armed if their work post is exterior to the protected area.

5. If search procedures of individuals and packages they may be carrying are not stipulated for all persons and hand-carried packages entering the protected area, then selection of individuals and hand-carried packages for search should be on the basis of a random process which is exercised each time an individual is about to enter the protected area.

6. Essential vehicles (Section 3.3.1) allowed access to protected areas include those designated strictly for security or emergency purposes, or vehicles not used primarily for conveyance of people that must be allowed within the protected area to serve a required function.

7. Picture badge identification should be used to satisfy Sections 3.3.2.1 and 3.3.2.2, with special color coding or symbols to satisfy 3.4.1, when inside vital areas.

8. Casual visitor groups, such as tour groups do not constitute "persons having a need to enter such (vital) areas", Section 3.4.1.

Implementation of the physical security program should be accomplished one to two months before fuel loading. Security features required for new fuel in storage prior to loading of the first unit should be implemented as of the time fuel is onsite.

III. REVIEW PROCEDURES

At the PSAR stage, the review consists of a careful examination of the information submitted and comparison with the acceptance criteria set forth in II above. The general plant description in Chapter 1 and site-related information in Chapter 2 of the PSAR should be examined to determine if there are unique features that should be considered in establishing the physical protection program. It may be desirable at this stage to discuss the formulation of this program with the applicant.

At the FSAR stage, the physical security plan is reviewed to determine its conformance with the regulations, the information requirements of I above, and the acceptance criteria of II above. Applicable regulations, the position statements in Regulatory Guide 1.17, and the requirements and recommendations of ANSI N18.17-1973 are used as check lists for this review. The reviewer may also use appropriate Division 5 Regulatory Guides to the extent they are applicable to physical protection programs at nuclear power plants. Those having potential applicability are listed in the references. It is particularly important that the reviewer assure himself that all

items of vital equipment are contained within vital areas. A site visit by the reviewer may be necessary, during the construction phase, before the evaluation of the plan can be completed.

IV. EVALUATION FINDINGS

The evaluation finding at the PSAR stage should be substantially equivalent to the following statement:

> "The applicant has provided a general description of plans for protecting the plant against potential acts of industrial sabotage. Provisions for the screening of employees at the plant, and for design phase review of plant layout and protection of vital equipment have been described and conform to Regulatory Guide 1.17. We conclude that the applicant's arrangements for protection of the plant against acts of industrial sabotage are satisfactory for this stage of the licensing process."

The evaluation finding at the FSAR stage should be substantially equivalent to the following statement:

> "The applicant has submitted a comprehensive physical security plan for the protection of the plant against potential acts of industrial sabotage. This plan has been withheld from public disclosure pursuant to 10 CFR 2.790(d).
>
> "This plan has been reviewed and found to contain features considered essential for such a program by the staff. In particular, it has been found to comply with the Commission's regulations including 10 CFR 50.34(c) and applicable sections of 10 CFR Part 73, and conforms to the positions set forth in Regulatory Guide 1.17."

V. REFERENCES

1. Regulatory Guide 1.17, Revision 1, "Protection of Nuclear Power Plants Against Industrial Sabotage."

2. ANSI N18.17-1973, "Industrial Security for Nuclear Power Plants," American National Standards Institute (1973).

3. Regulatory Guide 5.7, "Control of Personnel Access to Protected Areas, Vital Areas, and Material Access Areas."

4. Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials."

5. Regulatory Guide 5.20, "Training, Equipping, and Qualifying of Guards and Watchmen."

6. 10 CFR 50.34(c), "Physical Security Plan."

7. 10 CFR Part 73, "Physical Protection of Plants and Materials."

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SECTION 14.1 INITIAL TEST PROGRAM - PRELIMINARY SAFETY ANALYSIS REPORT

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - Reactor Systems Branch (RSB)
Instrumentation & Control Systems Branch (ICSB)

I. AREAS OF REVIEW

The QAB reviews the following areas relating to the initial plant test programs described in Chapter 14 of the preliminary safety analysis report (PSAR) submitted by the applicant as part of its construction permit (CP) application. Due to uncertainties in final design and lack of control by the designer over the initial test program of customers who reference the design, no review is made in this area at the preliminary design review stage for standard plant designs.

1. Scope of Test Program

The initial plant test program is normally divided into four major phases: preoperational tests; initial fuel loading and precritical tests; low power tests; and power ascension tests. The following descriptive information provided for each phase is reviewed: (1) the definition of each phase; (2) the general testing objectives for each phase and the general prerequisites to be completed before each phase is begun; (3) the extent to which the test program will be used to verify the adequacy of construction and design of both the nuclear portion of the facility and the balance-of-plant; (4) advance plans for organizations, including those of the applicant, nuclear steam supply system (NSSS) vendor, architect-engineer, and others that will participate in the development and execution of the test program, the general responsibilities of these organizations; and the interfaces between the organizations; (5) the applicant's planned involvement in the development and approval of test procedures, conduct of the tests, and review and approval of test results; (6) the extent that the applicant will use its NSSS vendor, architect-engineer, its own engineering units, or other qualified design personnel, to provide scoping documents containing testing objectives and acceptance criteria for use in developing detailed test procedures; (7) the minimum qualifications established for personnel involved in the approval of test procedures, conduct of tests and approval of test results; and (8) the plans for augmenting the plant staff during the initial test program.

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.

2. Plant Design Features that are Special, Unique, or First-of-a-Kind

The applicant's description of the preoperational and startup tests planned for principal plant design features that are special, unique, or first-of-a-kind is reviewed. The applicant's plans relative to special prototype in-plant functional testing requirements for such features are reviewed to establish the extent that such test requirements will verify design performance objectives.

3. Regulatory Guides and Industry Codes and Standards

The applicant's plans for utilizing applicable regulatory guides and industry codes and standards in the development of its test program are reviewed.

4. Utilization of Plant Operating and Testing Experience at Other Reactor Facilities

The applicant's methods for normal review and evaluation of plant operating experiences at other reactor facilities are reviewed.

The scope of the applicant's program, the applicant's schedule for conducting such studies, the organizational units responsible for conducting the study, their general qualifications, and the method by which the results of this study will be factored into the test program are reviewed.

5. Test Program Schedule

The applicant's schedule for developing and conducting the different phases of the test program is reviewed and is compared with the expected fuel loading date; to determine if adequate time has been allotted to develop and conduct a comprehensive preoperational test program; to establish that preoperational and startup test procedures will be available in adequate time; and to establish that plant operating and emergency procedures will be available in sufficient time to permit their trial use during the test program. The schedules are also reviewed to assure the availability of sufficient numbers of qualified plant and support (augmenting) personnel for the test program.

6. Trial Use of Plant Operating and Emergency Procedures

The applicant's plans for trial use of plant operating and emergency procedures during the test program are reviewed to determine the extent to which such procedures will be trial tested.

The Reactor Systems Branch reviews the application to establish if any of the systems within their area of review responsibility differ significantly from previously reviewed designs. If the designs differ significantly, RSB evaluates the impact of the changes on the initial test program with particular emphasis on the need for any special design features required to perform an acceptable initial test program.

The ICSB reviews the application to establish if digital based core protection calculators proposed for the facility differ significantly from previously

approved designs. If the design differs significantly, ICSB evaluates the impact of the changes on the initial test program with particular emphasis on the need for any special design features required to perform an acceptable initial test program.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described above are as follows:

1. Scope of Test Program

The primary objective of the review of information on the scope of the test program in the PSAR is to assure that the applicant recognizes the need to develop and conduct a comprehensive test program and that the applicant has performed the necessary early planning for successful achievement of this goal. Particular criteria that should be satisfied are as follows:

a. The description of the test program should identify and define the major phases of preoperational testing, fuel loading, precritical testing, low power testing, and power ascension testing. The definitions provided should be consistent with those included in Regulatory Guide 1.68 or justification should be provided for exceptions.

b. The applicant's stated test program objectives should provide assurance that the adequacy of construction, as well as design, will be verified for structures, systems, and components important to safety.

c. The applicant should designate the responsible organizations that will participate in the test program and there should be reasonable assurance that such designated organizations can collectively provide the necessary level of staffing with suitable skills and experience to develop and conduct the test program.

d. The applicant should plan to utilize the plant operating and technical staff in the development and conduct of the test program and in the review of test results.

e. The applicant should establish plans for the designers of structures, systems, and components, which may include the NSSS vendor, architect-engineer, or other design groups, to provide test objectives and acceptance criteria to be used in the development of detailed test procedures.

f. The applicant's program should provide assurance that construction of structures, systems, and components will be essentially complete prior to beginning preoperational testing.

g. The applicant's program should provide assurance that construction tests and inspections will be essentially completed before preoperational testing of the item or system is begun. Such testing or inspection may include system flushing and cleaning, wiring checks, leak tightness tests, initial calibration of instrumentation, and subsystem and component functional tests.

h. The applicant should describe the administrative control that will assure that preoperational test prerequisites will be satisfied.

i. The applicant should describe the plans for establishing minimum qualification requirements to assure that appropriately qualified personnel will be assigned to conduct preoperational and startup tests and review and approve test procedures and results.

2. Plant Design Features that are Special, Unique, or First-of-a-Kind

a. The applicant should identify all principal plant design features or systems that are special, unique, or first-of-a-kind.

b. The applicant's description of planned tests should establish that such design features and systems will be functionally tested to demonstrate performance requirements.

3. Regulatory Guides and Industry Codes and Standards

The applicant should commit to provide, in the FSAR, detailed plans relative to conforming to regulatory positions in regulatory guides applicable to testing.

4. Utilization of Plant Operating and Test Experiences at Other Reactor Facilities

The applicant should describe plans for routine evaluation of reactor operating experiences and describe how the results will be factored into the test program. The organizations or designated management positions responsible for this review should be qualified to conduct this activity. The schedule for conducting the review should be consistent with the schedule for development of the detailed test procedures. The types of reactor facilities that will be included in the review should include plants similar in design to the applicant's plant. The time frame included in the applicant's study should be sufficient to achieve a recent survey of plant operating problems. In general, this would involve the previous two-year period. The sources of information to be utilized should be adequate to conduct a meaningful and comprehensive review.

5. Test Program Schedule

a. The applicant should provide a schedule, relative to the expected fuel loading date, for the beginning and end of each major phase of the test program. The schedule for conducting preoperational testing should provide for at least nine months of actual testing.

b. The preoperational testing phase should be scheduled such that most of the plant operating and technical staff can participate in the development, conduct, and review of test results. The hiring and training schedules for these personnel, as presented in Chapter 13 of the PSAR, should be compatible with the preoperational testing schedule. In general, the hiring and training of such personnel should be planned for completion at least one year prior to the time of expected fuel loading.

c. The schedule for development of plant operating and emergency procedures should establish that they will be available for trial use during the test program.

6. Trial Use of Plant Operating and Emergency Procedures

The applicant's plans and commitments in this area should establish that plant operating and emergency procedures will be use-tested to the extent practicable, during the preoperational test phase.

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

Preparations for the review of Chapter 14 of the PSAR should include familiarization with the applicant's commitments in Chapter 13 pertaining to organization and staffing, plant procedures, and training. It should also include familiarization with the commitments in the SAR pertaining to transfer of plant structures, systems, and components from the vendor or constructors to the applicant when construction or installation is completed. The reviewer should also be familiar with the contents of Regulatory Guides 1.68 and 1.33.

The review consists of an analysis of the information submitted in Chapter 14 of the PSAR and a detailed comparison with each of the acceptance criteria contained in subsection II above. Coordination of the review with the assigned reviewers and/or project manager may be necessary for items 2, 5, and 6 listed in subsection I above.

Plant test programs are not rigidly fixed; however, experience has shown that certain elements are common to and necessary for all plants.

Each element of the SAR information is to be reviewed against this SRP section. The reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the magnitude and uniqueness of the project. Any exceptions or alternatives are to be carefully reviewed to assure that they are clearly defined and that an adequate basis exists for acceptance.

When the QAB reviewer has completed his review and has prepared round 1 and round 2 questions and SER, a copy of each should be formally transmitted to the secondary review branches (RSB and ICSB) for their review and input prior to issuance. The secondary review branches should normally be given 2 weeks to formally respond; however, shorter or longer schedules may be appropriate for a particular facility depending on the overall licensing review schedule. All requests for information or staff positions issued to the applicant on information in Chapter 14 are either to be issued by QAB or issuance coordinated with QAB to assure consistency in application of review criteria and that questions, staff positions, and the applicant's responses will appear in the appropriate section of the PSAR (423 series questions and responses on Chapter 14).

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been presented in the PSAR to support conclusions of the following type, to be included in the staff's safety evaluation report.

"The staff has completed its review of the information provided on the initial test program in support of the construction permit application for (facility name). The review was conducted in accordance with SRP Section 14.1 and included:

1. Evaluation of the scope of the applicant's test program including the responsibilities and qualifications of test personnel in participating organizations; the general testing objectives, the major phases of the test program, and the applicant's plans to augment the station staff, as necessary, during the test program;

2. Evaluation of the testing proposed for unique or first-of-a kind design features for the facility;

3. Evaluation of the applicant's plans for review and utilization, where applicable, of operating experiences from other reactors in the development of the test program;

4. Evaluation of the applicant's test program schedule to establish that sufficient time for testing is planned and that the schedule is compatible with the schedules for the hiring and training of plant personnel;

5. Evaluation of the applicant's plans to utilize plant operating and emergency procedures to the extent practicable during the preoperational testing phase.

On the basis of this review, the staff has concluded that assurance has been provided that the applicant has establisned acceptable plans for the initial test program. The staff concludes that the information provided satisfies the acceptance criteria in SRP Section 14.1. Additional review of the Initial Test Program will be conducted by the staff during the Operating License review."

V. REFERENCES

1. Regulatory Guide 1.8, "Personnel Selection and Training."

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SECTION 14.2 INITIAL PLANT TEST PROGRAM - FINAL SAFETY ANALYSIS REPORT

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - Analysis Branch (AB)
Core Performance Branch (CPB)
Instrumentation & Control Systems Branch (ICSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The QAB reviews items 1 through 11 below, relating to initial plant test programs, described in Chapter 14 of the final safety analysis report (FSAR) submitted by the applicant as part of its operating license (OL) application. For standard plant designs, see SRP Section 14.3, Initial Plant Test Program - Final Design Approval.

1. Summary of Test Program and Objectives

The summary descriptions for each major phase of the test program and the specific objectives for each major phase are reviewed.

2. Organization and Staffing

The information provided on the applicant's organizational units and any augmenting organizations, or other personnel that will manage or execute any phase of the test program, is reviewed. The information provided on the responsibilities, authorities, level of staffing, and qualifications of principal participants is reviewed. The information describing the extent and nature of the participation of the plant operating and technical staff in the test program is reviewed.

3. Test Procedures

The system the applicant will use to develop, review, and approve individual test procedures is reviewed. The responsibilities of the organizational units that will perform these activities, the designated functions of each organizational unit, and the general steps to be followed in conducting these activities are reviewed. The type and source of design performance information that will be, or is being, used in the development of detailed test procedures is reviewed. The format for the test procedures is also 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.

4. Conduct of Test Program

The administrative controls that will govern the conduct of each major phase of the test program are reviewed. The specific methods to be used to assure that prerequisites are satisfied for individual tests are reviewed. The procedures to be followed in initiating plant modifications, or repairs that are determined to be required by the test program are reviewed. The measures taken to ensure that necessary retesting is performed following repairs or modifications are reviewed. The controls that will be in effect to require adherence to approved test procedures are reviewed.

5. Review, Evaluation, and Approval of Test Results

The controls that will govern the review, evaluation, and approval of test results for each phase of the test program are reviewed, including the specific controls to be used to assure notification of responsible organizations, such as design organizations, when test acceptance criteria are not met and the specific controls established to resolve such problems. The applicant's controls relating to the methods and schedules for approval of test data for each major phase are reviewed as well as the methods used for initial review of parts of multiple tests (e.g., hot functional testing) prior to proceeding with the test phase.

6. Test Programs' Conformance with Regulatory Guides

The applicant's plans pertaining to conformance with regulatory guides applicable to initial test programs (see subsection V) are reviewed to establish the extent of conformance. Exceptions to regulatory positions in applicable regulatory guides are reviewed, along with the justification provided, for each guide.

7. Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program

The status of the applicant's program for review of operating experiences at other reactor facilities is reviewed. The degree of conformance to commitments in the preliminary safety analysis report (PSAR) is reviewed. A summary of the principal conclusions or findings from this review and the effect on the test program are reviewed.

8. Trial Use of Plant Operating and Emergency Procedures

The information pertaining to how, and to what extent, the plant operating, emergency, and surveillance procedures will be use-tested during the test program is reviewed.

9. Initial Fuel Loading and Initial Criticality

The procedures that will guide initial fuel loading and initial criticality, including the prerequisites and precautionary measures to be used to assure safety, are reviewed.

10. Test Program Schedule and Sequence

The schedule for conducting each phase of the testing program relative to fuel loading date, is reviewed. Information pertaining to anticipated schedule overlap of the test program with test program schedules for any other reactor facilities at the site is reviewed. The anticipated sequential test schedule for preoperationally testing individual plant structures, systems, and components for each test phase is reviewed. This review is conducted to assure (1) that the planned sequence of testing will provide for appropriate prerequisite system tests prior to conducting tests of critical systems or items and (2) that the approximate scheduled time for testing (relative to fuel loading) of critical structures, systems, and components is established such that reasonable confidence can be maintained in test results in the time period between the tests and fuel loading (during which time construction and other test activities are being conducted). The sequence for conducting the tests planned for the startup test phase is reviewed. The time available between approval of testing procedures and their intended use is reviewed.

11. Individual Test Descriptions

The individual test abstracts for each test phase are reviewed to establish the degree of conformance with applicable tests identified in Regulatory Guide 1.68, other applicable regulatory guides, and other special testing requirements for the facility, including those identified for special or unique plant features. The objectives for each test, summary of prerequisites and test method, and specific acceptance criteria for each test are reviewed to establish that the functional adequacy of those structures, systems, or components involved will be verified.

The AB reviews information in Chapter 14 of the FSAR relating to those startup transient tests that will be used, in part, for validation of transient codes and design models that were used for predicting plant performance. Particular emphasis is given to test objectives, test methods, and acceptance criteria relating to transient code and design model verification.

The ICSB and AB review the testing proposed for on-line digital based protection calculators to assure that the proposed test methods and acceptance criteria are adequate to verify the performance of the system, both during the preoperational test phase and the startup test phase.

The CPB reviews the information in Chapter 14 relating to core physics tests to assure that the test objectives, test methods, and the acceptance criteria are acceptable and consistent with the design requirements for the facility.

The RSB reviews the information in Chapter 14 relating to the items listed below to assure that the test objectives, test methods, and the acceptance criteria are acceptable and consistent with the design requirements for the facility:

1. Decay Heat Removal System

2. Reactor Core Isolation Cooling System (BWRs)

3. Primary Pressure Relief System

4. Leak Detection System

5. Emergency Core Cooling System

6. Natural Circulation Test (PWRs)

7. Loss of Reactor Coolant System Flow Test

8. Control Rod Scram Test

9. Transient Tests

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described above are as follows:

1. Summary of Test Program and Objectives

The applicant's description should establish that the major phases of the program and the objectives for each phase are consistent with the general guidelines and applicable regulatory positions contained in Regulatory Guide 1.68 or the justification provided for any exceptions should be found to be acceptable by the reviewer.

2. Organization and Staffing

The information provided should conform to the general guidelines and regulatory positions contained in Regulatory Guides 1.68 and 1.33. The minimum qualification requirements of principal participants are found to be acceptable based on the reviewer's judgment. Factors to consider in this judgment are the assigned responsibilities of augmenting personnel and the overall organizational plan for the test program.

The applicant's commitments, pertaining to how and to what extent the plant operating and technical staff will be utilized, should provide assurance that such personnel will be used to the maximum extent practicable. Participants in the test program should receive plant-specific training/indoctrination in the administrative controls for the test program prior to the start of testing. The level of staffing should be adequate based on the reviewer's judgment.

3. Test Procedures

The applicant's system that will be used to develop, review, and approve individual test procedures should provide for appropriate levels of review prior to final

approval. The different organizational units or personnel performing these functions should possess, by virtue of prior training and experience, the necessary qualifications to develop comprehensive and technically sound test procedures. The format for the test procedures should be similar to the format contained in Regulatory Guide 1.68 or the justification for exceptions should be found to be acceptable by the reviewer. The applicant should utilize system designers to provide the test objectives and acceptance criteria used in developing detailed test procedures. The participating system designers should include those of the nuclear steam supply system (NSSS) vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as appropriate.

4. Conduct of Test Program

   a. The test program should be conducted by appropriately qualified personnel using detailed procedures approved by designated management positions within the applicant's organization.

   b. The controls used by the applicant to assure that test prerequisites are met should include requirements for inspections, checks, etc., require identification of test personnel completing data forms or checksheets and require identification of dates of completion.

   c. The controls provided for plant modification and repairs, identified as a result of plant testing, are found to be acceptable if: (a) the controls are sufficient to assure the required repairs or modifications will be made, (b) the controls will assure retesting is conducted following such modifications or repairs, and (c) the controls will assure a review of any proposed facility modifications by the original design organization or other designated design organizations. The applicant's requirements for documentation associated with such controls should permit audits to be made to assure proper implementation of controls.

   d. The controls pertaining to adherence to test procedures and to methods for changing test procedures are found to be acceptable based on the reviewer's judgment. Modifications to startup test procedures should be made in accordance with technical specifications for post-fuel loading tests.

5. Review, Evaluation, and Approval of Test Results

   a. The controls that will govern the review, evaluation, and approval of test results should provide for a technical evaluation of test results by qualified personnel and approval of test results by designated management positions in the applicant's organization.

   b. Design organizations should be notified and should participate in the resolution of problems involving design that result in or contribute to a failure to meet test acceptance criteria.

c. The applicant should establish requirements for review and approval of the test data for each major test phase, prior to beginning the next phase.

d. The applicant should establish requirements for review and approval of the test data at each major power test plateau or power/flow test condition before proceeding to the next test level during the power ascension test phase.

6. Test Program's Conformance with Regulatory Guides

The applicant should establish and describe an initial test program that is consistent with regulatory positions in applicable regulatory guides listed in Subsection V. If exceptions to regulatory positions are taken, the applicant should specifically identify all exceptions, and provide suitable justification. Exceptions to regulatory positions will be reviewed for acceptability on a case-by-case basis. The applicant should address all applicable regulatory guides and identify them by guide number and revision number.

7. Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program

a. The study conducted by the applicant of operating experiences should have been accomplished substantially in accordance with PSAR commitments.

b. The conclusions reached, and the effect on the test program, should recognize and account for categories of reportable occurrences that are repeatedly being experienced and other operating experiences of safety concern.

8. Trial Use of Plant Operating and Emergency Procedures

The applicant should incorporate the plant operating, emergency, and surveillance procedures into the test program or otherwise verify these procedures through use to the extent practicable during the test program.

9. Initial Fuel Loading and Initial Criticality

The procedures that will guide initial fuel loading and initial criticality should include precautions, prerequisites, and measures consistent with the guidelines and regulatory positions contained in Regulatory Guide 1.68 or exceptions should be found to be acceptable by the reviewer.

10. Test Program Schedule and Sequence

a. At least nine months should be allowed for conducting preoperational testing.

b. At least three months should be allowed for conducting startup testing including fuel loading, low power tests, and power ascension tests.

c. Overlapping test program schedules (for multi-unit sites) should not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.

d. The approximate sequence and time schedule (relative to fuel loading) for conducting preoperational tests are deemed acceptable based on the reviewer's judgment.

e. The sequential schedule for individual startup tests should establish, insofar as practicable, that test requirements will be completed prior to exceeding 25% power for all plant structures, systems, and components that are relied upon to prevent, to limit or to mitigate the consequences of postulated accidents.

The schedule should also establish that, insofar as practicable, testing will be accomplished as early in the test program as feasible and that the safety of the plant will not be totally dependent on the performance of untested systems, components, or features.

f. Test procedures should be in a form suitable for review by regulatory inspectors at least 60 days prior to their intended use, or for fuel loading and startup test procedures, at least 60 days prior to fuel loading.

11. Individual Test Descriptions/Abstracts

The applicant should provide abstracts of planned tests for structures, systems, components, and design features that will provide assurance of testing of those items that meet any of the criteria listed below. Abstracts should be provided for those structures, systems, components or design features that:

a. Will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period; or

b. Will be used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions; or

c. Will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications; or

d. Are classified as engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits; or

e. Are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the FSAR; or

f. Will be used to process, store, control, measure, or limit the release of radioactive materials.

The abstracts should include objectives, prerequisites, test methods, significant parameters and plant performance characteristics to be monitored, and acceptance criteria in sufficient detail to establish that the functional adequacy of structures, systems, components, and design features will be demonstrated by tests.

If the method for testing of a structure, system, or component will not subject the item or system under test to representative design operating conditions, the test abstract should contain sufficient information to justify the test method to be used.

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

Preparation for the QAB review of Chapter 14 of the FSAR should include the following:

1. Review of the Commission's Safety Evaluation Report that was issued upon review of the Construction Permit application. This review is conducted for the purpose of familiarization with the principal design criteria for the facility and to identify any specific staff or ACRS concerns or unique design features that may warrant special test consideration.

2. Review of Chapters 1 through 12 of the FSAR for the purpose of familiarization with the facility design and the nomenclature that is applied to structures, systems, and components within the facility.

3. Review of Chapters 13 and 17 of the FSAR for the purpose of familiarization with the applicant's organizational structure, qualifications, administrative controls, and quality assurance program as these apply to or impact on the initial test program.

4. Review of Chapter 15 of the FSAR to identify assumptions pertaining to performance characteristics that can and should be verified by testing and to identify all structures, systems, components, and design features that are assumed to function (either explicitly or implicitly) in the accident analysis that can and should be tested.

5. Review of a computer listing (or equivalent) of Licensee Event Reports for operating reactors of similar design to identify potentially serious events and chronic or generic problems that warrant special test consideration. Computerized information on Licensee Event Reports can be obtained through the Office of Management and Program Analysis.

6. Review of the Standard Technical Specifications for the type facility to identify all structures, systems, and components that will be relied upon for establishing conformance with safety limits or limiting conditions for operation.

7. The reviewer should be familiar with the results of startup tests for reactors of similar design (reported to the Commission in Startup Test Reports) to assure that problem areas will be appropriately factored into the review of the initial test program.

8. The reviewer should be familiar with regulatory guides that are applicable to initial test programs.

The review consists of comparison of the information in Chapter 14 or, if applicable, other chapters in the FSAR with the acceptance criteria provided in subsection II above. Each element of the SAR information is to be reviewed against this SRP section. Although plant test programs are not rigidly fixed, experience has shown that certain elements are common to and necessary for all plants. The reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the magnitude and uniqueness of the project. The reviewer should recognize that certain structures, systems, and components are more important to safety than others. The review of individual tests should be conducted in a graded manner to assure that the most emphasis is placed on structures, systems, and components that are considered to be engineered safety features. Any exceptions or alternatives are to be carefully reviewed to assure that they are clearly defined and that an adequate basis exists for acceptance. Coordination of the review with the assigned reviewer for Chapter 13 may be necessary for items 2 and 10 with the project manager and other branches for item 11 above as deemed necessary by the reviewer. In general, the appropriate branch will be requested by the QAB to review proposed testing for special, unique, or first-of-a-kind design features to establish the adequacy of proposed testing including the validity of test methods. The reviewer is responsible for review and evaluation of all subsequent amendments to the FSAR up to the time the Operating License is issued to assure that any changes in design or commitments that impact on the initial test program continue to satisfy the acceptance criteria described in subsection II above.

When the QAB reviewer has completed preparation of requests for additional information, staff positions, and SER, they should be formally transmitted to the assigned secondary review branches for their review and comments prior to issuance. Under normal conditions a period of 2 weeks should be provided for their review and comment; however, longer or shorter schedules may be in order depending on the overall licensing schedule.

IV. EVALUATION FINDINGS

When the review of the information in the FSAR is complete and the reviewer has determined that it is satisfactory and in accordance with the acceptance criteria in subsection II above, a statement of the following type should be provided for the staff's Safety Evaluation Report:

"The staff has reviewed the information provided in the Final Safety Analysis Report on the applicant's test program in accordance with SRP Section 14.2. This review included an evaluation of: (1) The applicant's organization and staffing for the development, conduct, and evaluation of the test program; (2) the qualifications and experience of the principal participants managing and supervising the test program; (2) the administrative controls that will govern the development, conduct, and evaluation of the test program; (4) the participation of the plant operating and technical staff in the test program; (5) the applicant's requirements pertaining to the trial-use of plant operating and emergency procedures during the test program; (6) the schedule for conducting the test program; (7) the sequence of testing to be followed; (8) the methods for conducting individual tests and the acceptance criteria to be used in evaluating the test results for plant structures, systems, and components; and (9) the test programs' conformance with applicable regulatory guides including (provide list, guide no., revision no., and date). The review also included an evaluation of the results of the applicant's review of reactor plant operating experiences, conducted to determine where improvement or emphasis was warranted in the initial test program. The staff has concluded that the information provided in the application meets the acceptance criteria in Section 14.2 of the SRP and describes an acceptable initial test program that will demonstrate the functional adequacy of plant structures, systems, and components."

V. REFERENCES

1. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

2. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

3. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants."

4. Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

6. Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments."

7. 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."

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

9. Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."

10. Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems."

11. Regulatory Guide 1.108, "Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants."

12. Regulatory Guide 1.128, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants."

13. Regulatory Guide 1.8, "Personnel Selection and Training."

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SECTION 14.3 STANDARD PLANT DESIGNS, INITIAL TEST PROGRAM - FINAL DESIGN APPROVAL (FDA)

REVIEW RESPONSIBILITIES*

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

The QAB reviews the following areas relating to the initial plant test program described in Chapter 14 of the Standard Safety Analysis Report (SSAR) submitted by the applicant (NSSS vendor or architect-engineer) as part of its application for final design approval for a standard plant:

1. Participation in Test Program

The applicant's (designer) availability to participate in the development of the test program is reviewed including the applicant's plans to develop scoping documents containing testing objectives and acceptance criteria for use in developing detailed test procedures.

2. Plant Design Features that are Special, Unique, or First-of-a-Kind

The applicant's description of preoperational or startup tests recommended for principal plant design features that are special, unique, or first-of-a-kind is reviewed. The applicant's recommendations relative to special prototype in-plant functional testing requirements for such features are reviewed to establish the extent that such test requirements will verify design performance objectives.

3. Regulatory Guides

Alternative testing proposed for structures, systems, components, and design features within the scope of design where testing is not feasible or that will not permit testing in accordance with regulatory positions in applicable regulatory guides is reviewed. Test abstracts for other structures, systems, and components within the scope of design 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.

II. ACCEPTANCE CRITERIA

1. Participation in Test Program

The applicant should commit to make available to the customer its services for participation in the initial test program. As a minimum, the applicant should plan to: (1) submit scoping documents containing testing objectives and acceptance criteria to the customer for use in developing detailed test procedures under its scope of design responsibility, and (2) participate in the resolution of problems involving or impacting on design that are disclosed by failure of tests to meet acceptance criteria.

2. Plant Design Features that are Special, Unique, or First-of-a-Kind

The applicant should identify all principal plant design features or systems that are special, unique, or first-of-a-kind. The applicant's description of recommended tests should establish that such design features or systems can be functionally tested to demonstrate performance requirements.

3. Regulatory Guide

The applicant should identify all plant features that are not compatible with testing in full compliance with regulatory guides applicable to initial test programs. Regulatory guides which should be considered are those listed in subsection V of SRP Section 14.3. Alternative testing recommended for plant features, so identified, will require evaluation on a case-by-case basis. Abstracts of other recommended tests within the scope of design should be in accordance with regulatory positions in applicable regulatory guides.

III. REVIEW PROCEDURES

The review consists of familiarization with the design as described in the SSAR and review of the information submitted in Chapter 14 of the SSAR. Coordination of the review with the assigned reviewers in the Division of Systems Safety may be necessary for Items 1, 2, and 3 listed in subsection II above. Plant test programs are not rigidly fixed; however, experience has shown that certain elements are common to and necessary for all plants. Each element of the SSAR information is to be reviewed against this SRP section. Reviewer's judgment during the review is to be based on an inspection of the material presented, whether items of special safety significance are involved, and the magnitude and uniqueness of the project. Any exceptions or alternatives are to be carefully reviewed to assure tht they are clearly defined and that an adequate basis exists for acceptance.

When the reviewer has determined that each of the acceptance criteria listed above has been satisfied, based upon information in the SSAR, the review is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been presented in the SSAR to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant has committed to make available its services for participation in initial test programs of customers referencing this design and has stated that the design for this portion of the plant is such that testing may be accomplished, to the extent practicable, in accordance with applicable regulatory guides. In cases where the design will not permit testing in accordance with applicable regulatory guides, the applicant has proposed acceptable alternative testing for recommendation to its customers. Because the applicant has no control over the administration of the test program, because of possible different interactions between this portion of the plant and the remainder of the plant, and because of possible evolvement in standard technical specifications, the initial test program will be reviewed in accordance with SRP Sections 14.1 or 14.2 (as appropriate) during the construction permit and operating license review stages of each application that references this standard design."

V. REFERENCES

1. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

2. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

3. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants."

4. Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

6. Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments."

7. 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."

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SECTION 15.0 INTRODUCTION

The review branches in the Division of Systems Safety and the Division of Site Safety and Environmental Analysis that have some responsibility for review of the postulated transients and accidents listed in this chapter are as follows:

Reactor Systems Branch (RSB) - reviews analyses of the transients and accidents from the viewpoint of systems operation and core thermal-hydraulic performance.

Core Performance Branch (CPB) - reviews analyses of core physics and fuel behavior.

Analysis Branch (AB) - performs generic reviews of thermal-hydraulic computer codes and topical reports.

Effluent Treatment Systems Branch (ETSB) - reviews analyses of postulated spills of radioactive material outside containment.

Accident Analysis Branch (AAB) - evaluates possible radiological consequences of the transients and accidents.

Instrumentation and Controls Systems Branch (ICSB) - reviews and evaluates the reactor protection and safety instrumentation and control instrumentation.

Power Systems Branch (PSB) - reviews and evaluates emergency onsite power.

Auxiliary Systems Branch (ASB) - evaluates the operation and performance of the auxiliary systems following accidents and transients (i.e., auxiliary feedwater systems) and classification of structures, components and systems.

Mechanical Engineering Branch (MEB) - evaluates the mechanical design of the reactor primary system and components, pipe restraints and other safety-related mechanical components.

Containment Systems Branch (CSB) - evaluates the response of the containment to postulated accidents.

Events such as fires, floods, storms, or earthquakes are not explicitly considered in the review of postulated transients and accidents in Chapter 15. Rather, the requirements to design for these postulated events are reviewed under other SRP sections, as listed in Table 15-1 (attached).

The reviewers are responsible for the selection and emphasis of aspects of the reviews of transients and accidents in this chapter. Judgment on the areas to be given attention during each review is based on an inspection of the information provided, the similarity of this material to that recently reviewed on other plants, and whether items of special or unique safety significance are involved.

The Reactor Systems Branch reviews the majority of the postulated transients and accidents, as indicated in the SRP sections for Chapter 15. Various anticipated process disturbances,

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equipment malfunctions, and postulated component failures are examined to evaluate the plant capability to control or accommodate such failures and malfunctions. The RSB review includes an analysis of the pressure to which the reactor and steam systems are subjected and the need for engineered safety features to mitigate the consequences of accidents. The impact of various single failures on the course of a transient or accident is also considered.

The Core Performance Branch is responsible for the review of all reactor physics data presented in this chapter. This includes power levels, power distributions, Doppler coefficients, moderator temperature coefficients, void coefficients, reactor kinetics parameters, and control rod worths. The CPB review includes the evaluation of possible damage to the fuel. CPB also reviews the operating bands and uncertainty bands associated with these variables and assists RSB as requested.

The review of transients and accidents by RSB requires an evaluation of results, presented in the application, of analytical methods which frequently are not documented in the application. In such cases, the applicant may refer to a vendor topical report. The methods include DNB (departure from nucleate boiling) correlation development, subchannel analysis, system transient analysis, analysis of RIA (reactivity-initiated accidents), and LOCA (loss-of-coolant accident) analysis. For those cases where applicants use techniques previously considered and approved by the staff, additional review of methods may not be required. However, if new methods are involved, the AB performs a review of topical reports and other information which describe the method of analysis. Such a review generally includes vendor model description, data correlations and empirical relationships, solution techniques, summary of computer codes if involved, sample problems, experimental verification, and comparative calculations.

In its review of transients and accidents, RSB may request from AB, as the need is identified, an independent check of the results submitted by the applicant. In such cases, AB obtains input data from the applicant for use in auditing codes available to the staff.

Upon request of RSB, the Instrumentation and Control Systems Branch provides assistance in evaluating the sequence of postulated events, protective and safeguards systems actuation and potential bypass modes, and manual control. ICSB determines whether reactor protection and safeguards controls and instrumentation will function as assumed in the transient analysis with regard to manual or automatic actuation; remote sensing, indication, and control; and interlocks with auxiliary or shared systems. PSB determines the adequacy of onsite emergency power systems.

The AAB review is concentrated on those more severe accidents that could result in the release of radioactive materials and could have significant radiological consequences involving the general public. AAB determines the potential doses resulting from the accidents and compares these doses to established dose criteria and guidelines. Based on the results of these analyses, AAB determines the adequacy of equipment designed to mitigate radiological consequences. In addition, radiological analyses are made to determine certain technical specification limits for safety-related equipment and structures.

The ASB provides an evaluation of the auxiliary systems to confirm that these systems can supply all the functions required to support ECCS during and following a DBA LOCA or any other NSSS pipe break.

CSB evaluates the functional capability of the containment and subcompartments for the spectrum of loss-of-coolant events under SRP Section 6.2.1. CSB, on request from the RSB, also provides an evaluation of containment pressure calculations utilized in the reflood portion of the ECCS performance analyses.

MEB is responsible for the review of the effects of blowdown loads on core support structures and on control rod guide structures. MEB verifies that the core remains in place in case of a LOCA and that the control rods can be inserted. MEB is also responsible for evaluating the effects of blowdown loads including jet forces on the piping of the reactor coolant system and on the support structures of the components of the reactor coolant system. MEB verifies that acceptable criteria have been employed in the design of the reactor coolant system and its supports to prevent failures in the reactor coolant pressure boundary or in engineered safety feature equipment in the event of a LOCA.

TABLE 15-1

Postulated Event	SRP Section		Branches Having Primary Review Responsibility
Accidents at Nearby Locations	2.2.3	Evaluation of Potential Accidents	Accident Analysis Branch
Storms	2.3	Meteorology	Site Analysis Branch
	3.3	Wind and Tornado Loadings	Structural Engineering Branch
Floods	2.4	Hydrologic Engineering	Site Analysis Branch
	3.4	Water Level (Flood) Design	Auxiliary Systems Branch
Earthquakes	2.5	Geology and Seismology	Site Analysis Branch
	3.2	Classification of Structures, Components and Systems	Auxiliary Systems Branch
	3.7	Seismic Design of Structures	Structural Engineering Branch
	3.8	Design of Category I Structures	Structural Engineering Branch
	3.10	Seismic Design of Category I Instrumentation and Electrical Equipment	Mechanical Engineering Branch
	3.11	Environmental Design of Mechanical and Electrical Equipment	Instrumentation and Control Systems Branch Power Systems Branch
Missiles	3.5	Missile Protection	Accident Analysis Branch Reactor Systems Branch Auxiliary Systems Branch Structural Engineering Branch
Fires	9.5.1	Fire Protection System	Auxiliary Systems Branch

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SECTION 15.1.1
15.1.2
15.1.3
15.1.4

DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

A number of transients which are expected to occur with moderate frequency, and which involve an unplanned increase in heat removal by the secondary system, are covered by this review plan. Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. The power level increase will lead to a reactor trip. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure.

Each of the transients covered by this plan should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The transients to be evaluated include:

1. Pressurized Water Reactors (PWR's) and Boiling Water Reactors (BWR's)
   a. Feedwater system malfunctions that result in a decrease in feedwater temperature.
   b. Feedwater system malfunctions that result in an increase in feedwater flow.
   c. Steam pressure regulator malfunctions or failures that result in increased steam flow.
2. PWR's Only
   a. Inadvertent opening of a steam generator relief or safety valve.

The topics covered in the primary review include: postulated initial core and reactor conditions which are pertinent to feedwater system malfunctions, pressure regulator or

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safety or relief valve malfunctions, methods of thermal and hydraulic analysis, postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor system components, functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the transient analysis are reviewed to ensure that the values of pertinent system parameters are within the ranges expected for the type and class of reactor under review. The parameters include: peak clad temperature, peak fuel temperature, core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (MCHFR) or minimum critical power ratio (MCPR), departure from nucleate boiling ratio (DNBR), vessel water level, thermal power, vessel pressure, steam line pressure (for BWR's), steam line flow (for BWR's), feedwater flow (for BWR's), and reactivity.

The sequence of events described in the SAR for these transients is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrument and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients which result from an increase in heat removal are:

   a. To identify which of the moderate-frequency* transients that result in increased heat removal are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

*The term "moderate-frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 8 and 9.

a. Pressures in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 2).

b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 is satisfied throughout the transient.

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model should be suitably conservative. The use of the following values is considered acceptable:

   a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used for both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values are used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

RSB reviews the applicant's description of the transients caused by excessive heat removal with specific attention to the occurrences that lead to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that a particular transient involving an increase in heat removal is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the increase-in-heat-removal transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the SRP for Chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the values of reactivity coefficients and control rod worths used by the applicant in his analysis, and the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates the justification provided by the applicant to show that the core burnup selected yields the minimum margins. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variation with time during the transient of parameters listed in Sections 15.x.x.3(c) and 15.x.x.4(c) of the Standard Format (Ref. 1) is reviewed. The values of the more important of these parameters, as listed in Section I of this SRP, are compared to those predicted for other similar plants to see that they are within the range expected.

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:

"A number of plant transients can result in an unplanned increase in heat removal by the secondary system. Those that might be expected to occur with moderate frequency can be caused by feedwater system or pressure regulator malfunctions or the inadvertent opening of a steam generator safety or relief valve (PWR only). All of these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the _____transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)* did not decrease below _____and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with respect to transients resulting in an unplanned increase in heat removal by the secondary system that are expected to occur with moderate frequency."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

*Minimum critical heat flux ratio (MCHFR) or minimum critical power ratio (MCPR) for a BWR.

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.1.5 SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system and results in a reduction of coolant temperature and pressure. Due to the negative moderator temperature coefficient this cooldown causes an increase in core reactivity. The core reactivity increase causes a power level increase and a decrease in shutdown margin. If the plant is at power, the reactor is automatically tripped and the main steam and feedwater line isolation valves are automatically closed. Decay heat is removed through the unaffected steam generators by venting steam from the secondary system safety and relief valves. The auxiliary feedwater system supplies makeup water to the unaffected steam generators.

The transient following a steam line break is sensitive to the discharge rate so that a range of break sizes must be evaluated both inside and outside containment to determine the acceptability of the system response. The course the transient takes and its ultimate effects also depend on the assumed initial power level and mode of operation (i.e., hot shutdown, full power, one-, two-, or three-loop operation). Analyses with various assumed initial conditions are required to verify that the condition leading to the severest consequences has been identified.

The topics reviewed include: postulated initial core and reactor conditions pertinent to the steam line break accident, methods of thermal and hydraulic analyses, postulated sequence of events including analyses to determine the time of reactor trip and time delays prior to and subsequent to initiation of the reactor protection system, assumed responses of the reactor coolant and auxiliary systems, functional and operational characteristics of the

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.

reactor protection system in terms of its effects on the sequence of events, and operator actions required to secure and maintain the reactor in a safe shutdown condition.

The results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. The parameters of importance for these transients include reactor coolant system (RCS) pressure, steam generator pressure, fluid temperatures, fuel and clad temperatures, discharge flow rate, steam line and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, mass and energy transfer within the containment (for breaks inside containment), reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

APCSB reviews the auxiliary feedwater system to see that it can function following a steam line break given a single active component failure with either onsite or offsite power. This is done as described in Standard Review Plan (SRP) 10.4.9. RSB reviews the auxiliary feedwater system to see that the flow provided is acceptable for controlling the transient following a steam line break.

MEB evaluates potential water-hammer effects on safety valve integrity.

AAB evaluates the fission product release and verifies that the radiological consequences resulting from a steam line break are within acceptable limits. This evaluation is performed for the design basis case as described in the appendix to this review plan.

CSB, as described in SRP 6.2.1, evaluates the response of the containment to ruptures of steam lines with regard to the effects of pressure and temperature on the containment functional capabilities.

II. ACCEPTANCE CRITERIA

1. The general objective of the review of steam line rupture events is to confirm that the primary reactor coolant system is maintained in a safe status for a range of steam line ruptures up to and including a break equivalent in area to the double-ended rupture of the largest steam line.

2. The specific criteria against which the consequences of these breaks are to be evaluated are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above 1.30 or 1.32, as appropriate, based on correlations given in References 3 and 4. If the DNBR falls below these values, fuel damage (rod perforation) must be assumed unless it can be shown, based on an acceptable fuel damage model, that fuel failure has not occurred. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

   c. The radiological criteria used in the evaluation of steam system pipe break accidents (PWR's only) appear in the appendix to this plan.

3. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

   a. The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular nuclear steam supply system (NSSS) design, and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the SAR.

   b. Assumptions as to the loss of offsite power and the time of loss should be made so as to maximize the consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break, or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feedwater pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents.

c. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Positions APCSB 3-1 and MEB 3-1 (Ref. 8).

d. The worst single active component failure should be assumed to occur. The assumed single failure may cause more than one steam generator to blow down, or may be in any of the systems required to control the transient.

e. The maximum-worth rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used.

f. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

g. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results, but for the analysis of steam line break accidents this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

1. The reviewer determines the acceptability of the analytical models and assumptions as follows:

a. The values of system parameters and initial core and system conditions used as input to any analytical model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the analysis. The reviewer confirms that the amount of secondary coolant expelled from the system (for breaks outside containment) has been calculated conservatively by evaluating

the applicant's methods and assumptions, by comparing with an acceptable analysis performed on another plant of similar design, or by comparing with staff calculations for typical plants done by CPB on request.

b. The acceptability of the equations, sensitivity studies, and models proposed by the applicant are evaluated. For situations where new generic methods are proposed, the reviewer will request an evaluation by CPB. Acceptable equations, sensitivity studies, and models are described in References 5, 6, and 7.

c. Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements:

(1) Reactor trip signal: credit taken for any reactor trip signal is reviewed by EICSB to confirm that, under accident conditions, the instrumentation and control systems are capable of the assumed response.

(2) Emergency core cooling system (ECCS): credit taken for actuation of the ECCS is reviewed by EICSB to verify the ability of the instrumentation and control systems to respond as assumed.

(3) Auxiliary feedwater system: the availability of the auxiliary feedwater system to supply adequate auxiliary feedwater flow to the intact steam generators during the accident and the subsequent shutdown condition is evaluated. This is done by APCSB as to availability of the system and by RSB as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion.

d. The variations with time during the transient of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The values of the more important of these parameters for the steam line break accident (as listed in Section I of this SRP) are compared with those predicted for other similar plants to see that they are within the range expected.

2. To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. This phase of the review is done using the system review procedures described in the SRP for Chapters 5, 6, 7, 8, and 10 of the SAR. The reviewer also considers single failures that may cause more than one steam generator to blow down, thus increasing the reactivity addition to the core.

3. The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests for verifying that valve discharge rates and response times (including, for example, opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam

isolation valves, and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for auxiliary feedwater system actuation, safety injection signal delay time, and delay times for delivery of any high concentration boron solution required to bring the plant to a safe shutdown condition.

4. Based on the above information, AAB evaluates the radiological consequences of the design basis steam line break accident as described in the appendix to this plan.

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 analyses and effects of steam line break accidents inside and outside containment, during various modes of operation and with or without offsite power, have been reviewed. The accident which resulted in the most severe consequences was determined and evaluated using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the spectrum of steam line break accidents showed that fuel damage was minimal and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) experienced by any fuel rod was _____, resulting in __% of the rods experiencing clad perforation. The maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of _____between the water and steam phases and a X/Q value of _____sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table _____. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within the 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike."

"Based on the above, the staff concludes that the plant design is acceptable with regard to steam line break accidents."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution." Journal of Nuclear Energy, Vol. 21, 241-248 (1967).

4. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox Power Generation Systems, BAW-10000, Supplement 1, March 1971.

5. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

6. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

7. "Standard Nuclear Steam Supply System B-SAR-241," Babcock and Wilcox Company, February 1974 (under review).

8. Branch Technical Positions 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 System Piping Outside Containment," attached to SRP 3.6.2.

APPENDIX

STANDARD REVIEW PLAN SECTION 15.1.5

RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURES OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Hydrology and Meterology Branch (HMB)
Reactor Systems Branch (RSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The AAB reviews (a) the sequence of events in the applicant's description of the steam line failure accident outside containment, with and without offsite power, to assure that this sequence of events represents the most severe case from the standpoint of release of radioactive materials, and (b) the coolant activity concentration technical specification limits to assure that potential doses resulting from this accident are adequately limited. The HMB provides acceptable atmospheric dispersion X/Q values for this accident. The ETSB provides acceptable models and assumptions for iodine spiking and their effects upon coolant activity. The RSB determines the acceptability of the applicant's description of events, including operator actions, for this accident.

II. ACCEPTANCE CRITERIA

Standard Technical Specification (STS) limits on PWR primary and secondary coolant activity concentrations and primary-to-secondary leak rate have been issued (References 4, 5 and 6). The plant is considered adequately designed against the consequences of a main steam line failure outside containment if calculations show that the resulting doses at the exclusion area and low population zone boundaries, based on STS limits, are: (a) small fractions (less than 10%) of the 10 CFR Part 100 exposure guidelines, and (b) within 10 CFR Part 100 guidelines for the cases of a preaccident iodine spike or one rod held out of the core.

If the doses are not within these guidelines using the STS limits, the technical specification limits on coolant concentrations and/or primary-to-secondary leak rate are reduced until the calculated doses are within these guidelines.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix 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.

At the construction permit stage, there is generally insufficient information available to make meaningful radiological consequence calculations for these accidents. At this stage, the review is limited to a brief review of the applicant's discussion of the main steam line failure accidents to determine that there are no unusual design features that would preclude the limitation of radiological consequences by appropriate limits on coolant concentrations and primary-to-secondary leakage. The detailed review of radiological consequences of the main steam line failure accidents is done at the operating license stage when system parameters are fully developed.

The Standard Technical Specifications for each of the three PWR vendors' NSSS include limits on the primary and secondary coolant activities and primary-to-secondary leak rate. If the applicant proposes to use these standard limits and his plant is one of the standard NSSS/BOP plants for which the steam line failure accident has been evaluated generically with the standard coolant activity and leakage limits, the reviewer need not reevaluate the offsite doses from this accident if the X/Q's for the site under review are lower than the limiting X/Q used in the generic review of the standard plant steam line failure.

The AAB review of main steam line failure accidents at the operating license stage consists of the following:

1. Review of the applicant's descriptions of the steam line failure accident (with and without offsite power). This includes a review of the time sequence of occurrence of events.

2. The RSB should be contacted for a determination of the acceptability of the applicant's description of events including operator actions. The AAB reviews the sequence of events to assure that this sequence of events represents the most severe case from the standpoint of release of radioactive materials and calculated doses.

3. Determination of coolant activity concentrations. The reviewer assumes the primary and secondary coolant activity concentrations allowed by the technical specifications (SAR Chapter 16 or the Standard Technical Specifications given in References 4, 5 or 6) as equilibrium conditions prior to the accident.

4. Determination of the iodine spiking effects. Two cases are analyzed, one with an iodine spike assumed to begin because of reactor trip or primary system depressurization when the steam line break occurs, and one with an iodine spike assumed to have begun well before the accident due to a previous reactor transient (Refs. 3 and 7).

The ETSB provides the AAB with the iodine spiking model. If the applicant proposes to use a spiking model different than the model described below, the ETSB should be requested to review the model.

The current spiking model assumes that at the time the steam line break occurs, the iodine release rate from the fuel rods to the primary coolant (expressed in unit of curies per unit time) increases by a factor of 500 relative to the release rate calculated assuming that the primary coolant iodine concentration is at the equilibrium concentration technical specification limit.

For the case with an iodine spike which already exists (due to a previous transient), the iodine concentration is assumed to be that allowed by Figure 3.4-1 of References 4, 5 or 6, or that which is proposed in Chapter 16 of the SAR.

5. Review of the effects of possible fuel damage during the accident on exclusion area boundary and LPZ doses. Additional coolant activity may become available for release if fuel failures result from the accident. The RSB reviews the effect of a steam line break (with the most reactive control rod stuck at its fully withdrawn position) on core thermal margins. If this event is predicted to cause fuel failures, RSB notifies AAB. The reviewer assumes the applicant's calculations of fuel damage are correct unless informed otherwise by RSB. If fuel damage does occur, calculations should be performed in order to assure that 10 CFR Part 100 guidelines are not exceeded (without a preaccident iodine spike).

6. Determination of the leakage into the steam generators. Normal operating primary-to-secondary leakage is assumed to exist in the steam generators. The leakage rate should be the maximum allowed by the technical specifications. This value is 1 gpm in the STS but may be lower if required because of the consequences of a rod ejection accident or an anticipated transient without scram (ATWS). The leakage should be apportioned between affected and unaffected steam generator(s) in such a manner that the calculated dose is maximized.

7. Determination of iodine transport. During periods of steam generator dry-out, all iodine transported to the secondary side by primary coolant leakage is assumed to be released to the environment. During periods of total submergence of the tubes, the fraction of iodine lost is equal to the flash fraction of the primary coolant leakage; appropriate credit for scrubbing by the secondary coolant may also be claimed (Reference 2). Any iodine transferred to the secondary side coolant will become airborne at a rate which is a function of the steaming rate and iodine partition coefficient. An iodine partition coefficient of 100 between steam generator water and steam phases may be conservatively assumed, unless the applicant presents reasonable evidence that the use of some other value is justified.

8. The AAB provides the HMB with the release location and release conditions. The HMB then provides the AAB with appropriate X/Q values for this accident.

9. Calculation of the exclusion area boundary (EAB) and low population zone (LPZ) boundary doses. The reviewer computes the doses for the steam line break accident, both with and without preaccident iodine spiking. A breathing rate of 3.47×10^{-4} m^3/sec is used in the calculation of thyroid doses for the first 8 hours following the steam line break.

10. Review of the results of the dose calculations. The calculated doses are considered acceptably low if: (1) the doses calculated without assuming the existence of a pre-accident iodine spike, but assuming that iodine spiking occurs as a result of the accident, are less than a small fraction of 10 CFR Part 100 guidelines; and (2) the doses calculated assuming preaccident iodine spiking or additional fuel failures occur as a result of the accident (assuming the most reactive control rod remains stuck in its fully withdrawn position) are less than the dose guidelines of 10 CFR Part 100. If the doses are not within these guidelines, the technical specification limits on equilibrium and/or spiked primary coolant activity, or the technical specification limit on primary-to-secondary leak rate, should be reduced until the calculated doses are within these guidelines.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings in the staff's evaluation report at the operating license stage:

> "The radioactivity release has been evaluated according to a conservative description of the plant response to the accident. Our calculated doses are presented in Table ____ and our assumptions are listed in Table ____.
>
> "Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within the 10 CFR Part 100 exposure guidelines even if the accident should occur with a preaccident iodine spike or assuming additional fuel failures occur during the accident as a result of the most reactive control rod remaining fully withdrawn."

The following paragraph is added if fuel damage is found to be a possible consequence of the accident:

> "The evaluation of the main steam line failure outside containment has been evaluated with ____% fuel damage in the core (as a result of the most reactive control rod remaining fully withdrawn). The resulting doses are within the guidelines of 10 CFR Part 100 provided the normal operating primary-to-secondary leakage is limited to _____ gpm."

At the construction permit stage, the following paragraph is included with the RSB findings in the staff's safety evaluation report:

"On the basis of our experience with the evaluation of steam line and steam generator tube failure accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations and/or primary to secondary leak rates so that potential offsite doses are small. At the operating license stage, we will include appropriate limits on these parameters to be included in the plant technical specifications."

If the plant is a standard plant which has been reviewed before, the following paragraph may be used:

"The radiological consequences of a steam line break in the (vendor's name) standard NSSS/(A/E's name) BOP were analyzed generically by the staff and the results reported in the staff SER dated (NSSS/BOP SER issue date). The offsite doses were found to be acceptable for sites with X/Q's equal to or less than _____ sec/m^3, if the coolant activities and steam generator primary-to-secondary leak rate are limited to the values in the (vendor's name) Standard Technical Specifications (NUREG-_____, (issue date)). The technical specifications the applicant will use include these coolant activity and steam generator leak limits. The staff has estimated the 0-2 hour X/Q at the exclusion area boundary of the site is _____ sec/m^3, as discussed in Section (2.3.4) in this SER. This X/Q is lower than the limiting X/Q for this standard NSSS/BOP plant. Therefore, we conclude that the offsite doses from a steam line break accident in the _____ plant would be less than the current dose guidelines and are acceptable, although a specific dose calculation for this accident has not been performed."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. A. K. Postma and P. S. Tam, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, USNRC, 1978.

3. R. R. Bellamy, "A Regulatory Viewpoint of Iodine Spiking During Reactor Transients," Trans. Am. Nucl. Soc., 28 (1978).

4. Standard Technical Specifications for Combustion Engineering PWRs, NUREG-0212.

5. Standard Technical Specifications for Westinghouse PWRs, NUREG-0452.

6. Standard Technical Specifications for Babcock and Wilcox PWRs, NUREG-0103.

7. W. F. Pasedag, "Iodine Spiking in BWR and PWR Coolant Systems," CONF-770708, 3-217 (1977).

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SECTION 15.2.1	LOSS OF EXTERNAL LOAD, TURBINE TRIP,
15.2.2	LOSS OF CONDENSER VACUUM, CLOSURE OF
15.2.3	MAIN STEAM ISOLATION VALVE (BWR),
15.2.4	AND STEAM PRESSURE REGULATOR
15.2.5	FAILURE (CLOSED)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

A number of transients which are expected to occur with moderate frequency result in an unplanned decrease in heat removal by the secondary system. These are covered in this review plan. Each of the transients should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The transients to be evaluated are:

1. Loss of External Load

In a loss of external load event an electrical disturbance causes loss of a significant portion of the generator load. This loss of load situation is different from the loss of a-c power condition considered in Standard Review Plan (SRP) 15.2.6 in that offsite a-c power remains available to operate the station auxiliaries (such as reactor coolant pumps). The onsite emergency diesels are therefore not required for the loss of external load transient. Immediate fast closure of the turbine control valves (TCV) and intercept valves is initiated whenever a loss of generator load takes place. For a boiling water reactor (BWR), a fast TCV closure (0.150-0.2 sec) causes a sudden reduction in steam flow and results in a reactor pressure surge. For a BWR without select rod insert (SRI), reactor scram occurs. For a pressurized water reactor (PWR) there is also a sudden reduction in steam flow, and this causes the pressure and temperature in the shell side of the steam generator to increase. The latter effect, in turn, results in an increase in reactor coolant temperature, a decrease in coolant density, an increase in water volume in the pressurizer, and an increase in reactor coolant pressure. For a PWR with an integrated control system, reactor power can be run back to a lower level on TCV closure.

In all light water-cooled reactors, sensible and decay heat can be removed through actuation of one or several of the following systems: steam relief system, steam

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bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling systems, and auxiliary feedwater system (PWR).

2. Turbine Trip

In a turbine trip event a malfunction of a turbine or reactor system causes the turbine to be tripped off the line by abruptly stopping steam flow to the turbine. This is different from the loss of electrical load condition described above in that fast closure of the turbine stop valves (TSV) is initiated. The TSV have faster (0.1 sec) closure times than the turbine control valves, resulting in more severe transients. For typical BWR and PWR plants, position switches on the TSV sense the trip and initiate reactor scram. The remainder of this transient is similar to the previously discussed loss of electrical load.

3. Loss of Condenser Vacuum

A loss of condenser vacuum event is one of the malfunctions that can cause a turbine trip. The remarks in 2, above, thus apply to this transient.

4. Main Steam Isolation Valve Closure

The main steam isolation valve (MSIV) transient for BWR's can be initiated by various steam line or reactor system malfunctions and by various operator actions. As the MSIV's close, position switches initiate a reactor scram when the valves in three or more of the steam lines are less than 90% open, the reactor pressure is above 600 psi, and the reactor mode switch is in the RUN position. The effect of MSIV closure is to limit steam flow to the turbine. The results are similar to those discussed in 1, above, but tend to be less severe since the MSIV closure time is much longer than that of the TCV.

5. Steam Pressure Regulator Failure

Steam pressure regulator failure in a closed position yields a transient similar to the transients discussed above. Generally, because the rate of change of system parameters is slower for a steam pressure regulator failure, a less severe transient results.

The review of the transients described above includes the sequence of events, the analytical models, the values of parameters used in the analytical models, and the predicted consequences of the transients.

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequences described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient analyses are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the analyses are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients listed in Section I are:

   a. To identify which of the moderate-frequency transients that result in an unplanned decrease in secondary system heat removal are the most limiting. (The term "moderate frequency" is used in this review plan in the same sense as in the definitions of design and plant process conditions in References 8 and 9.)

   b. To verify that, for the most limiting transients, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant should analyze these transients using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are

acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:

a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of these transients presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that any one of these transients is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the reduction-of-heat-removal transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic

initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variation with time of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the limiting transient are compared to those predicted for other similar plants to verify that they are within expected range.

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 (SER):

> "A number of plant transients can result in an unplanned decrease in heat removal by the secondary system. Those that might be expected to occur with moderate frequency are turbine trip, loss of external load, steam pressure regulator malfunctions, main steam isolation valve closure (in BWR's), loss of condenser vacuum, loss of non-emergency a-c power to the station auxiliaries, and loss of normal feedwater flow.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the _____ transient showed that cladding integrity was maintained by ensuring that the maximum departure from nucleate boiling ratio (or minimum critical heat ratio for a BWR) did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.
>
> "The staff concludes that the plant design is acceptable with regard to transients resulting in an unplanned decrease in heat removal by the secondary system that are expected to occur with moderate frequency."

*The SER should present one statement for moderate frequency transients involving an unplanned decrease in heat removal by the secondary system. Thus, the results of the reviews under SRP 15.2.6 and 15.2.7 are included in this statement.

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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SECTION 15.2.6 LOSS OF NON-EMERGENCY A-C POWER TO THE STATION AUXILIARIES

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

The loss of non-emergency a-c power is assumed to result in the loss of all power to the station auxiliaries. This situation could result either from a complete loss of the external grid (offsite) or a loss of the onsite a-c distribution system. It is different from the loss of load condition considered in Standard Review Plan (SRP) 15.2.2 because, in the latter case, a-c power remains available to operate the station auxiliaries. The major difference is that in the loss of a-c power transient all the reactor coolant circulation pumps are simultaneously tripped by the initiating event. This causes a flow coastdown as well as a decrease in heat removal by the secondary system.

Within a few seconds the turbine trips and the reactor coolant system is isolated, causing the pressure and temperature of the coolant to increase. A reactor trip is initiated by the turbine trip. For this transient, the diesel generators are automatically started and provide electric power to the vital loads. The sensible and decay heat loads are handled by actuation of the steam relief system, steam bypass to the condenser, reactor core isolation cooling system in a boiling water reactor (BWR), emergency core cooling system, and auxiliary feedwater system in a pressurized water reactor (PWR).

The review of the loss of a-c power transient includes the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote

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sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient analysis are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objective in reviewing the loss of a-c power transient is to confirm that either of the following criteria are met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of heat removal by the secondary system and has the same anticipated frequency classification.

   b. The plant responds to the loss of a-c power transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency* are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

*The term "moderate frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 8 and 9.

3. The applicant's analysis of the loss of a-c power transient should be based on an acceptable model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If the applicant proposes other analytical methods, these are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model should be suitably conservative. Use of the following parameter values is considered acceptable:

a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during the review of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of a-c power transient presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. The operation of standby diesel generators that is required.

If the SAR states that the loss of a-c power transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If

a quantitative analysis of the loss of a-c power transient is presented in the SAR, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, standby diesel generator and other systems needed to limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. This aspect of the review uses the procedures described in standard review plans for chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of parameters listed in 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the loss of a-c power transient are compared to those predicted for other similar plants to verify that they are within the expected range.

IV. EVALUATION FINDINGS

The evaluation findings under this plan are incorporated in a statement covering all transients of moderate frequency involving a decrease in heat removal by the secondary system. See the findings statement in SRP 15.2.1-5 for a typical statement.

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

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SECTION 15.2.7 LOSS OF NORMAL FEEDWATER FLOW

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power. The sequence of events for the loss of feedwater transient differs between a boiling water reactor (BWR) and a pressurized water reactor (PWR). A PWR has a backup (auxiliary and emergency) feedwater system while a BWR relies on the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system for backup core cooling. In either case, loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage.

For both PWR's and BWR's, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow. This can be accomplished by actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling system, and auxiliary feedwater system (PWR).

The review of the loss of feedwater transient includes the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

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The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the predicted results of the transient are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objective in the review of the loss of normal feedwater transient is to confirm that one of the following criteria are met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of heat removal by the secondary system, and has the same anticipated frequency classification.

   b. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency* are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

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*The term "moderate frequency" is used in this review plan in the same sense as in the definitions of design and plant process conditions in References 8 and 9.

3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If the applicant proposes to use other analytical methods, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model should be suitably conservative. The use of the following values is considered acceptable:

   a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during the reviews of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of normal feedwater flow transient presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. The operation of auxiliary systems that is required.

If the SAR states that the loss of feedwater transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of feedwater transient is presented in the SAR, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the loss of feedwater transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. This part of the review uses the procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding maximum pressure in the reactor coolant and main steam systems. The variation with time during the transient of parameters listed in Section 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the loss of normal feedwater transient are compared to those predicted for other similar plants to see that they are within the range expected.

IV. EVALUATION FINDINGS

The evaluation findings under this plan are incorporated in a statement covering all transients of moderate frequency involving an unplanned decrease in heat removal by the secondary system. SRP 15.2.1-5 contains a general statement covering the findings of this plan.

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Over-pressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.2.8 FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

The transient that results from a postulated feedwater line break is sensitive to the discharge rate; consequently, a range of break sizes should be evaluated both inside and outside containment to determine the acceptability of the response. Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup (by reducing feedwater flow to the affected steam generator). Therefore, analyses of various postulated break sizes and locations are needed to identify the particular situation that is most limiting with respect to system effects.

If a feedwater line rupture causes the water in the steam generator to be discharged through the break, the water will not be available for decay heat removal after reactor scram. The break location and size may be such to prevent addition of any feedwater to the affected steam generator. An auxiliary feedwater system is therefore provided to assure that feedwater is available to provide decay heat removal.

The review includes evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break, the methods of thermal and hydraulic analysis, the postulated sequence of events including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system actions, the assumed response of the reactor coolant and auxiliary systems, the functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition. The results of the analyses are reviewed to ensure that the values of

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.

pertinent system parameters, discussed in Section II below, are within expected ranges. The parameters of importance for these transients include reactor coolant system pressure, steam generator pressure, fluid temperatures, fuel and clad temperatures, discharge flow rate, steam line and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, mass and energy transfer within the containment (for breaks inside containment), reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

APCSB reviews the auxiliary feedwater system to verify that it can function following a feedwater line break, given a single active component failure and with either onsite or offsite power. This review is performed as described in Standard Review Plan (SRP) 10.4.9.

RSB reviews the auxiliary feedwater system to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break.

MEB evaluates potential water-hammer effects on safety valve integrity.

AAB evaluates the fission product release assumptions used in determining any offsite releases. AAB verifies that the radiological consequences resulting from a feedwater pipe break are within acceptable limits. This analysis is done independently of the RSB review.

CSB, under SRP 6.2.1, evaluates the response of the containment to breaks of feedwater lines with regard to the effects of pressure and temperature on the containment functional capabilities.

II. ACCEPTANCE CRITERIA

1. The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for a range of feedwater line breaks up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

2. The specific criteria used in evaluating the consequences of these breaks are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 3).

   b. The potential for core damage should be evaluated on the basis that it is acceptable if the minimum DNBR remains above 1.30 or 1.32, as appropriate, based on correlations given in References 4 and 5. If the DNBR falls below these values, fuel damage (rod perforation) should be assumed unless it can be shown, based on an acceptable fuel damage model, that no fuel failure results. If fuel damage is calculated to occur, it should be of sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

   c. Any activity release must be such that the calculated doses at the site boundary are well within the guidelines of 10 CFR Part 100.

3. There are certain assumptions which should be used in the analysis regarding important parameters that describe initial plant conditions and postulated system failures. These are listed below.

   a. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular nuclear steam supply system and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced if considered applicable.

   b. The assumptions as to whether offsite power is lost and the time of loss should be made conservatively. Offsite power may be lost simultaneously with the occurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should be made to determine the most conservative assumption appropriate to the plant design being reviewed. The study should take account of the effects that loss of offsite power has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater, and the resulting modification of the sequence of events.

   c. The effects of the postulated feedwater line breaks on other systems (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) should be considered in a manner consistent with the intent of Branch Technical Positions APCSB 3-1 and MEB 3-1 (Ref. 9).

d. The worst single active component failure should be assumed to have occurred in the systems required to control the transient.

e. The maximum rod worth should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be assumed.

f. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

g. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results for the case of a feedwater line rupture inside containment. However, this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.

III. REVIEW PROCEDURES

The procedures below are used during reviews of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB and are compared to the initial conditions listed in Section II of this plan. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of reactivity parameters used in the applicant's analysis.

Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The equations, sensitivity studies, and models proposed by the applicant are reviewed by RSB to determine if these have been previously reviewed and found acceptable by the staff. For situations where new generic methods are proposed, the reviewer will request an evaluation by CPB. Acceptable equations, sensitivity studies, and models are described in References 6, 7, and 8.

Credit taken for a reactor trip signal or for actuation of engineered safety features should be reviewed by EICSB to determine the ability of the instrumentation and control systems to respond as assumed under accident conditions.

The ability of the auxiliary feedwater system to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated by APCSB as to availability and by RSB as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to determine the maximum times permitted for their completion.

To the extent considered necessary, the RSB reviewer evaluates the effect of single active failures of systems and components that may alter the course of the accident. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, 8, and 10 of the SAR. The variations with time during the transient of parameters listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 2) are reviewed. The more important of these parameters for the feedwater line break accident (as listed in Section I of this SRP) are compared to those predicted for other similar plants to see that they are within the expected range.

The reviewer confirms that the amount of secondary coolant expelled from the system has been calculated conservatively by evaluating the applicant's methods and assumptions, by comparison with an acceptable analysis performed on another plant of similar design, or by comparison with staff calculations for typical plants which will be available from CPB on request.

The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests to verify that valve discharge rates and response times including, for example, opening and closing times (delay times) for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves, have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for actuation of the auxiliary feedwater system, safety injection signal delay time, and delay times for delivery of any high concentration boron injection required to bring the plant to a safe shutdown condition.

Using the information developed in the review, the AAB reviewer evaluates the radiological consequences of the design basis feedwater line break. This evaluation based on a qualitative comparison with the results of the design basis steam line break, or on a detailed analysis using the approach described in the appendix to Standard Review Plan 15.1.5.

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 analyses and effects of a spectrum of feedwater line breaks inside and outside containment, during various modes of operation, and with or without offsite power have been reviewed. The accident which resulted in the most severe consequences was determined and evaluated using a mathematical model that had been previously reviewed and found acceptable by the staff. The values of the parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analyses of the spectrum of feedwater line break accidents showed that fuel damage was minimal and that no loss of core cooling capability would result. The minimum departure from nucleate boiling ratio experienced by any fuel rod was _____, resulting in_____ of the rods experiencing cladding perforation. The maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The radiological consequences of the design basis feedwater line break have been evaluated. Technical specification limits on primary and secondary coolant activities limit potential doses to small fractions of 10 CFR Part 100 exposure guidelines.

"Based on the above, the staff concludes that the plant design is acceptable with regard to feedwater line break accidents."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

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

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

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, 241-248 (1967).

5. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox Power Generation Systems, BAW-10000, Supplement 1, March 1971.

6. "Reference Safety Analysis Report RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

7. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

8. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

9. Branch Technical Positions 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 System Piping Outside Containment," attached to SRP 3.6.2.

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SECTION 15.3.1
15.3.2

LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP AND FLOW CONTROLLER MALFUNCTIONS

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. The resulting increase in clad temperature could result in fuel damage. A number of transients that are expected to occur with moderate frequency and that result in a decrease in forced reactor coolant flow rate are covered by this review plan. Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Ref. 1).

Core thermal and hydraulic transients associated with partial and complete loss of reactor coolant flow are evaluated. These include:

1. For boiling water reactors (BWR's), partial and complete recirculation pump trips and malfunctions of the recirculation flow controller to cause decreasing flow.

2. For pressurized water reactors (PWR's), partial and complete reactor coolant pump trips.

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump, a fault in the power supply to the pump, a pump trip caused by such anomalies as over-current or phase imbalance, or a failure within the recirculation flow control network (BWR) resulting in decreasing flow. A complete loss of forced coolant flow may result from the simultaneous loss of electrical power to all pumps.

The review includes the postulated initial core and reactor conditions which are pertinent to the loss of flow transient, the methods of thermal and hydraulic analysis, the postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor systems components, the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

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.

The results of the applicant's flow transient analyses are reviewed to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. These parameters include: peak clad temperature, peak fuel temperature, core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (or minimum critical power ratio), departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), steam line flow (BWR), and feedwater flow (BWR).

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety system, and operator action to secure and maintain the reactor in a safe condition. The EICSB review, as described in Standard Review Plans (SRP) 7.2 and 7.3, concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. CPB, as described in the appendix to SRP 4.4, performs generic reviews of the thermal-hydraulic computer models used for this transient. CPB also performs, upon request, additional analyses related to these accidents for selected reactor types.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the RSB engineer to contact his counterpart in CPB to ensure that the relevant physics data have been used in any staff calculations.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of loss of forced reactor coolant flow transients are:

   a. To identify which of the transients are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds to the loss of flow transients in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency* are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures. (Ref. 2).

* The term "moderate frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 7 and 8.

b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 is satisfied throughout the transient.

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of the loss of reactor coolant flow transients should use an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of parameters used in the analytical model should be suitably conservative. The use of the following values is considered acceptable:

a. The reactor is initially at rated output (licensed core thermal power) for the number of loops assumed operating, plus 2% to account for power measurement uncertainty.

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

II. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of each of the loss of reactor coolant flow transients presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that are required.

5. The extent to which operator actions are required.

If the SAR states that a particular loss of flow transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The reviewer confirms that all types of flow loss transients are considered, e.g., pump trips during two-, three- and four-loop operation. The applicant is to present a quantitative analysis in the SAR of the loss of flow transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the loss of flow. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operation action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by the RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the applicant's proposed model.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of parameters listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the loss of reactor coolant flow transients are compared to those predicted for other similar plants to verify that they are within the expected range.

The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests to verity flow coastdown calculations.

V. 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 (SER):

"Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. The ones expected to occur during the life of the plant are those caused by reactor coolant (or recirculation) pump trips or a flow controller malfunction.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable by the staff. The values of the parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the _____ transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)** did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with regard to transients that are expected to occur during plant life and result in a decrease in reactor coolant flow rate."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review.)

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December, 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

*The SER should present one statement for all similar transients.
**Minimum critical heat flux ratio (MCHFR) or minimum critical power ratio (MCPR) for a BWR.

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

8. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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SECTION 15.3.3
15.3.4

REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Core Performance Branch (CPB)
Analysis Branch (AB)

I. AREAS OF REVIEW

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump in a pressurized water reactor (PWR) or recirculation pump in a boiling water reactor (BWR). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. This SRP section is intended to cover both of these infrequent transients.

The review is concerned with the postulated initial and long-term core and reactor conditions that are pertinent to the rotor seizure or broken shaft events, the methods of thermal and hydraulic analysis, the postulated sequence of events including time delays prior to and after protective system actuation, the assumed reactions of reactor system components, the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The parameters include: peak clad temperature, peak fuel temperature, core flow and flow distribution (including hydraulic instabilities), channel heat flux (average and hot), minimum critical heat flux ratio or critical power ratio, departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), steam line flow (BWR), and feedwater flow (BWR).

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.

The sequence of events described in the SAR is reviewed by both RSB and ICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The ICSB reviewer, as described in SRP Sections 7.2 and 7.3, concentrates on the instrumentation and controls aspects of the sequence described in the SAR and evaluates whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. ICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by AB. AB, as described in SRP Section 4.4, performs generic reviews of the thermal-hydraulic computer models used for this transient. AB also performs, upon request, additional analyses related to these accidents for selected reactor types.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the RSB engineer to contact his counterpart in CPB to ensure that the relevant physics data have been used in any staff calculations.

AAB is notified regarding the extent of fuel failures that are predicted by the analysis. AAB then evaluates the radiological consequences.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients resulting from a rotor seizure or shaft break in a reactor coolant pump are:

   a. To identify which of these infrequent transients is the more limiting.

   b. To verify that, for the more limiting transient, the plant responds in such a way that the criteria regarding fuel damage, radiological consequences, and system pressure are met.

2. The specific criteria for the rotor seizure and shaft break transients are:

   a. For events such as the rotor seizure or shaft break in a reactor coolant pump, the plant should be designed to limit the release of radioactive material to assure that doses to persons offsite are kept to values which are a small fraction of 10 CFR Part 100 guidelines.

   b. The potential for core damage should be evaluated on the basis of the acceptance criterion for DNBR in SRP Section 4.4 (Ref. 9). If DNBR falls below the limits of this criterion, fuel damage (rod perforation) should be assumed unless it

can be shown, based on an acceptable fuel damage model, that no fuel failure results. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

c. Pressure in the reactor coolant system and main steam system should be maintained below 110% of the design pressure (Ref. 1).

d. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

e. Only safety-grade equipment should be used to mitigate the consequences of the accident. Safety functions should be accomplished assuming the worst single failure of a safety system active component (see Refs. 2 and 3).

f. The ability to achieve long-term coolability of the core should be verified.

g. This event should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps.

3. The applicant's analyses should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 4 through 8 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by AB.

The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model.

a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

b. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core, and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

4. 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 both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicants' analyses of the rotor seizure and shaft break events are reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent and time at which operator actions are required.

If the SAR states that one of the transients is not as limiting as the other, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the transient that is determined to be more limiting. For the transient that is found more limiting, the reviewer confirms that the effects of the transient are determined for each mode of operation (e.g., one-, two-, three-, or four-loop) allowed by the technical specifications. Either a separate analysis should be presented for each mode of operation or the effects of each mode should be referenced to the limiting case.

For the more limiting transient, the RSB reviewer, with the aid of the ICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the transient.

This phase of the review uses the system review procedures described in the SRP sections for Chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, AB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria in subsection II regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR), core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), temperature (maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy), steamline pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system (if applicable) are reviewed. The more important of these parameters (as listed in section I of this SRP) are compared to those predicted for other similar plants to confirm that they are within the expected range. In particular, the peak cladding temperature and percentage of fuel rods that experience a departure from nucleate boiling condition are compared and AAB is notified regarding the extent of fuel failures predicted by the analysis.

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 analyses and effects of an instantaneous seizure of a rotor and an instantaneous break of a shaft of a reactor coolant pump* during any allowed mode of operation have been reviewed. It was found that the more limiting of these events was the _____. This event was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis showed that ___% of the fuel rods experienced

*Recirculation pump shaft for a BWR.

departure from nucleate boiling (DNB) and that the peak clad temperature reached was ___°F. This assures that the fuel damage will be minimal and that no loss of core cooling capability will result. The analysis showed that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with regard to a possible seizure of a rotor or break of a shaft of a reactor coolant pump."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

2. "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 1), U.S. Nuclear Regulatory Commission, NUREG-0138, November 1976.

3. "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 22), U.S. Nuclear Regulatory Commission, NUREG-0158, December 1976.

4. F. M. Bordelon, "Calculation of Flow Coastdown after Loss of Reactor Coolant Pump," WCAP-7973, Westinghouse Electric Corporation, August 1970.

5. C. D. Morgan, H. C. Cheatwood, and J. R. Glandermans, "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069, Babcock and Wilcox Company, July 1973.

6. R. H. Stoudt and J. E. Busby, "CADD - Computer Applications to Direct Simulation of Transient Events on Water Reactors," BAW-10080 (nonproprietary) and BAW-10076 (proprietary), Babcock and Wilcox Company, July 1973.

7. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

8. R. Linford, "Analytical Methods of Transient Evaluations in the General Electric Boiling Water Reactor," NEDP-10802, General Electric Company, April 1973.

9. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
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SECTION 15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The CPB evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor; and a single rod, with current control modes, for a boiling water reactor*) from a subcritical or low power (e.g., startup range) condition to assure conformance with the requirements of General Design Criteria 20 and 25 under this SRP section. The review under this SRP section covers the description of the causes of the transient and the transient itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transient as compared with the acceptance criteria. The reactivity coefficients and control rod worths utilized in this review are also evaluated by the CPB under SRP Section 4.3.

II. ACCEPTANCE CRITERIA

1. The following General Design Criteria (Ref. 1) apply:

   a. Criterion 25, which requires that the reactor protection system be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

   b. Criterion 20, which requires that the protection system action be initiated automatically.

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. Examples of limits used previously to satisfy this criterion are:

*If a single-failure proof system prevents the uncontrolled withdrawal of a control rod, this transient need not be considered for BWRs.

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.

(1) BWRs - The value of the minimum critical power ratio (CPR) calculated with the GETAB analysis (Ref. 2) will vary for different plants and product lines. A value of 1.06 is typical.

(2) PWRs - The value of the minimum departure from nucleate boiling ratio (DNBR) calculated with due allowance for mixing grids is typically 1.30 using the BAW-2 correlation (Ref. 3) or the W-3 correlation (Ref. 4). Depending upon the data base and design approach, other values are also acceptable.

If the application under review does not use one of these limits, then the correlation used by the applicant is reviewed by the Analysis Branch under SRP Section 4.4; a criterion for critical heat flux is thereby established that is utilized by the CPB for the review under this SRP section.

b. Fuel temperature and clad strain limits consistent with the acceptance criteria of SRP Section 4.2 should not be exceeded. For steady-state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate (usually expressed in kW/ft). Examples of this criterion are:

(1) For BWRs, a linear heat generation rate of 24-28 kW/ft, which would result in limited UO_2 melting, but is not sufficient to cause 1% clad strain and potential clad failure.

(2) For PWRs, a linear heat generation rate of 20-22 kW/ft, which would result in a centerline fuel temperature equal to or less than the melting point of UO_2.

The specific value of the linear heat generation rate for this criterion is established during each review in a manner consistent with the acceptance criteria of SRP Section 4.2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. REVIEW PROCEDURES

The procedures below are used for both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values are used in the analysis and the reviewer should compare these to the limiting safety settings included in the proposed technical specifications.

The reviewer, in determining whether the acceptance criteria are met, considers the following:

1. Peak conditions for the transient are maximized by low initial power; thus, the power level of the reactor should be at the lowest possible value compatible with

the control rod configuration used for the accident. The postulated initial reactor coolant flow, pressure and inlet temperature (i.e., the extremes of postulated conditions) should be consistent with the rod and power configuration to give minimum DNBR, or CPR conditions.

2. Peak conditions for the transient are maximized by large reactivity addition rates near prompt critical; thus, the control rod configurations for the assumed withdrawal must be examined to confirm that such a maximized state has been included in the calculations. For a PWR, control bank withdrawal should be used. For a BWR, with the present control rod withdrawal procedures, a single rod of maximum worth available in a normal configuration should be used. In many cases this will be a rod near the 50 percent rod density configuration. (More recent modes of BWR control such as group withdrawal may require that other configurations be examined.) The withdrawal rate should be the maximum available to the system.

3. The exact analysis of the transient would normally involve a three-dimensional, coupled neutron kinetics-thermal hydraulics calculation. However, acceptable results may be obtained with a neutron point-kinetics analysis and a coupled or separate hot fuel rod thermal analysis, if conservative input data are used. The reviewer determines whether the applicant's analytical methods are acceptable by using one or more of the following procedures:

   a. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports and reports prepared in response to technical assistance requests.

   b. Perform a de novo review of the method (usually described in a separate licensing topical report, and frequently handled outside the scope of the review for a particular facility).

   c. Perform auditing-type calculations with methods available to the staff.

   d. Require additional, bounding calculations by the applicant to cover portions of the applicant's analytical methods that have not been fully reviewed or approved.

4. The input to the neutron kinetics analysis model should be examined to assure that the input is appropriately conservative both for the state of the reactor and for the particular way it is used in the analysis. The power distribution or peaking factors used in the neutron kinetics and hot pin thermal calculations must provide a conservative representation of the control rod configuration under consideration. The Doppler feedback coefficient should be related conservatively to the values accepted in the review under SRP Section 4.3, considering the time in cycle and temperature conditions of the fuel. The use of beginning of lifetime (BOL) Doppler coefficients is the most conservative; they should be used unless other values are

specifically justified. Non-weighting of the coefficients is conservative, but weighting factors for the particular flux distribution shapes involved in the transients may be used if fully explored and justified. The moderator coefficients used should also be conservatively related to the values accepted in the review under SRP Section 4.3. The most positive or least negative values should be used and for a PWR this occurs at BOL. If the coefficient is negative, it may be conservatively taken as zero.

5. The analysis should consider the relationships between the particular spatial flux shapes for the transient and the nuclear instrument response to assure that scrams occur at the times used in the analysis, that valid scram power levels are assumed, and that conservative scram delays and reactivity functions are used.

6. The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, reactor temperatures and pressures, maximum heat flux levels, and the related fuel temperatures, DNBR, or CPR, and maximum clad strain. The latter are compared to the acceptance criteria.

IV. EVALUATION FINDINGS

If the staff, on completion of the review finds the applicant's analysis acceptable, conclusions of the following type should be included in the staff's safety evaluation report:

"The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods under low power startup conditions have been reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worths, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions and reactivity feedback effects due to moderator and fuel temperature changes, have been examined. (If check calculations have been done, they should be summarized.)

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert criteria from SRP Sections 4.2 and 4.4). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that the applicant's analyses of the maximum transients for single error control rod withdrawal from a subcritical or low power condition have been confirmed, that the analytical methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both assumptions and models, to assure that fuel damage will not result from such control rod assembly accidents."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions" and General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).

3. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

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

SECTION 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The CPB evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor; and a single rod, with current control modes, for a boiling water reactor) at power to assure conformance with the requirements of General Design Criteria 20 and 25 under this SRP section. The review under this SRP section covers the description of the causes of the transient and of the transient itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transients as compared with the acceptance criteria. The reactivity coefficients and control rod assembly worths involved are reviewed by the CPB under SRP Section 4.3.

II. ACCEPTANCE CRITERIA

1. The following General Design Criteria (Ref. 1) apply:

   a. Criterion 25, which requires that the reactor protection system be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

   b. Criterion 20, which requires that the protection system action be initiated automatically.

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. Examples of limits used previously to satisfy this criterion are:

      (1) BWRs - The value of the minimum critical power ratio (CPR) calculated with the GETAB analysis (Ref. 2) will vary for different plants and/or product lines. A value of 1.06 is typical.

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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) PWRs - The value of the minimum departure from nucleate boiling ratio (DNBR) calculated with due allowance for mixing grids is typically 1.30 using the BAW-2 correlation (Ref. 3) or the W-3 correlation (Ref. 4). Depending upon the data base and design approach, other values are also acceptable.

If the application under review does not use one of these limits, then the correlation used by the applicant is reviewed by the Analysis Branch under SRP Section 4.4; a criterion for critical heat flux is thereby established that is utilized by the CPB for the review under this SRP section.

b. Fuel temperature and fuel clad strain limits consistent with the acceptance criteria of SRP Section 4.2 should not be exceeded. For steady-state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate (usually expressed in kW/ft). Examples of this criterion are:

(1) For BWRs, a linear heat generation rate of 24-28 kW/ft, which would result in limited UO_2 melting, but is not sufficient to cause 1% clad strain and potential clad failure.

(2) For PWRs, a linear heat generation rate of 20-22 kW/ft, which would result in a centerline fuel temperature equal to or less than the melting point of UO_2.

The specific value of linear heat generation rate for this criterion is established during each review in a manner consistent with the acceptance criteria of SRP Section 4.2. For nonequilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. REVIEW PROCEDURES

1. The review process and the areas examined differ somewhat depending on whether a BWR or PWR is being reviewed. For both systems, the review covers the entire power range from low to full power and the allowed extreme range of reactor conditions during the operating (fuel) cycle including rod configurations, power distribution, and associated reactivity feedback components. The continuous withdrawal of normal configurations of rods should be assumed for the initial conditions in the transient calculation. For a PWR, this is one or two control banks; for a BWR, with current modes of control, it is a single control rod (future modifications under consideration may change this to group movement). The review covers a full range of rod or bank withdrawals up to maximum rod or bank worths and rates of reactivity addition.

The exact analysis of the transient would normally involve a three-dimensional, coupled neutron kinetics, thermal-hydraulics calculation. However, acceptable results may be obtained with suitable approximate calculations. The problem examined and the approximations used differ for a PWR and a BWR.

2. For a BWR, past analyses and reviews have shown that at maximum rod worths and rates of reactivity addition, the reactor power increases slowly and the total increase is relatively small, so that the transient may be approximated by steady-state analyses. Because of changes in local power distribution attributable to rod motion and strong void feedback effects on the power distribution, three-dimensional, steady-state, coupled neutron distribution, thermal hydraulics calculations that take account of these effects are required. The transient is halted by action of the rod block monitor (RBM) system, which should block rod withdrawal before fuel safety limits are reached.

   The review process for a BWR, while recognizing the inherent transient nature of the problem, is concentrated on the steady-state aspects of the transient to assure that initial and subsequent power distributions are maximized, that the reactor conditions produce minimum critical power ratio, CPR, and that the response of the RBM system is conservatively calculated considering minimum operation of the associated local power range monitoring system.

3. A PWR analysis, on the other hand, generally involves larger power changes and requires transient calculations. Because power distributions in the course of the transient can frequently be predicted conservatively using design-limit peaking factors, point kinetics may be used for the nuclear transient. The nuclear transient is coupled, however, to core and system thermal-hydraulic response to the power changes (fuel and moderator thermal feedback and system instrumentation response).

   For a PWR, the reviewer ascertains that a full range of transient conditions are explored, that the transient calculation models are adequate, and that scram response of the flux, temperature, or pressure instrumentation is correctly calculated. The range of parameters to be considered includes:

   a. Initial power levels from low to full power.

   b. Reactivity insertion rates from very low to maximum possible for the control system, including allowance for uncertainties.

   c. Fuel and moderator feedback reactivity coefficients covering the range expected throughout the cycle, including allowance for uncertainties.

   d. Power peaking factors at design limits for the initial power level conditions.

4. For both types of reactors the reviewer determines whether the applicant's analytical methods and models are acceptable, including steady-state, transient, system response, and fuel response models. This may be done by using one or more of the following procedures:

a. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports (SERs) and reports prepared in response to technical assistance requests.

b. Perform a de novo review of the method (usually described in a separate licensing topical report and frequently handled outside the scope of the review for a particular facility).

c. Perform auditing-type calculations with methods available to the staff.

d. Require additional, bounding calculations by the applicant to cover portions of the applicant's analytical methods that are not fully reviewed or approved.

5. The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod; scram or rod block actions that occur; reactor temperatures and pressures; maximum heat flux levels; and the related fuel temperatures, DNBR or CPR, and maximum clad strain. The latter are compared to the acceptance criteria.

IV. EVALUATION FINDINGS

If the staff, on completion of the review finds the applicant's analysis acceptable, conclusions of the following type should be included in the staff's safety evaluation report:

"The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transient and the instrumentation response to the transient. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined. (If check calculations have been done, they should be summarized.)

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert acceptance criteria from SRP Sections 4.2 and 4.4, as appropriate). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that the applicant's analysis of maximum transients for single error control rod malfunctions have been confirmed, that the analytical methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both input assumptions and models to assure that fuel damage will not result from control rod withdrawal transients."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions" and General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).

3. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

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

SECTION 15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

CPB reviews the following:

1. The types of control rod misoperations that are assumed to occur. For a pressurized water reactor (PWR), this may include one or more rods moving or displaced from normal or allowed control bank positions (such as dropped rods and rods left behind when inserting or withdrawing banks, or single rod withdrawal) and may include the automatic control system attempting to maintain full power. For a boiling water reactor (BWR) with current modes of control rod operation, limiting anomalies are reviewed under SRP Sections 15.4.1 and 15.4.2, and no additional areas are considered here.

2. Descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations.

   ICSB in SRP Sections 7.2 and 7.7 reviews those safety systems required to prevent misoperations, as required by General Design Criterion 25 (Ref. 2), as well as the control rod system. The purpose of the review is to determine what events are to be included as single error malfunctions (e.g., single rod withdrawal).

3. Descriptions of the sequence of events occurring during each transient, e.g., rod drop followed by automatic return to full power with possible power overshoot, including the effect of important feedback mechanisms and trips.

4. Descriptions of the calculational models used and justification of their validity and adequacy.

5. The input to the calculations, including rod worths, power distributions, and feedback coefficients, and evidence of the conservatism of the input.

USNRC STANDARD REVIEW PLAN

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6. Results of the analyses including, for each of the transients considered, plots of the time history of reactor power, reactor vessel pressure, critical heat flux for the limiting fuel rod, and maximum fuel centerline temperature or linear heat generation rate.

II. ACCEPTANCE CRITERIA

1. The following General Design Criteria apply:

   a. Criterion 25, which requires that the reactor protection system be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system (Ref. 2).

   b. Criterion 20, which requires that the protection system action be initiated automatically (Ref. 1).

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. An example of limits used previously to satisfy this criterion is that the minimum departure from nucleate boiling ratio (DNBR) should not be less than 1.30, using the W-3 correlation (Ref. 3), or using the BAW-2 correlation (Ref. 4). If the application under review does not use one of these limits, then the reviewer must assure that an acceptable correlation is established under SRP Section 4.4 and is used here.

   b. Fuel temperature and fuel clad strain limits consistent with the acceptance criteria of SRP Section 4.2 should not be exceeded. For steady-state or nearly steady-state conditions, this can be referred to a linear heat generation rate (usually expressed in kW/ft). An example of this criterion is a linear heat generation rate of 20 to 22 kw/ft, which would result in a centerline fuel temperature equal or less than the melting point of UO_2. The specific value of linear heat generation rate for this criterion is established in a manner consistent with the acceptance criteria of SRP Section 4.2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. REVIEW PROCEDURES

The reviewer, in determining whether the criteria are met, must determine the transients that should be considered for this event. Generally, the list of errors should include: inadvertently withdrawing one or several rods; leaving one or several rods behind during bank withdrawal; and inserting one or several rods with power compensation in other portions of the core. In addition to these events, the reviewer must also decide, by postulating single failures in equipment or errors in operation, whether additional

single rod malfunctions can be created. Once the list of transients has been established, the reviewer must determine acceptability in accordance with the criteria of subsection II.

1. For each failure event analyzed, the cases which result in a limiting fuel rod condition should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. Conditions of first order importance for any time in cycle are initial power level and distribution, initial rod configuration, reactivity addition rate, moderator temperature, fuel temperature and void reactivity coefficients.

2. For each event, the analytical methods used by the applicant are reviewed. Those steady-state and transient methods that are primarily based on reactor physics considerations are the responsibility of CPB. Where thermal-hydraulic methods are involved, assistance of the Analysis Branch may be requested as described in SRP Section 4.4. In either case, the reviewer must determine whether the applicant's evaluation methods are acceptable. This may be done by using one or more of the following procedures:

   a. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports (SERs) and reports prepared in response to specific technical assistance requests.

   b. Perform a de novo review of the method (usually described in a separate licensing topical report, and often handled outside the scope of the review for a particular facility).

   c. Perform auditing-type calculations with methods available to the staff.

   d. Require additional bounding calculations by the applicant to confirm the validity of those portions of the applicant's analytical method that have not already been fully reviewed and approved.

3. For each event, the results are evaluated. In addition to verifying conformance to the acceptance criteria of subsection II above, the reviewer determines that:

   a. Input conditions (e.g., pressure, temperature, flow rate) are at the adverse end of the range of values specified as the operating range.

   b. Initial power is 102% of licensed core thermal power, unless a lower power level is justified by the applicant.

   c. Output signals (power, temperature, flux perturbation) provided adequate alarm or scram signals.

   d. Nuclear conditions that interact with this event (e.g., Doppler coefficient, void coefficient) have been calculated as described in SRP Section 4.3.

IV. EVALUATION FINDINGS

If the reviewer's evaluation shows that the applicant's analyses are acceptable, the following kinds of statements should be included in the staff's safety evaluation report:

"The possibilities for single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects due to moderator and fuel temperature changes, have been examined. (If check calculations have been done, they should be summarized.)

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling (DNB) have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert criteria from SRP Sections 4.2. and 4.4). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that maximum configurations and transients for single error control rod malfunctions have been analyzed, that the analysis methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, in both input assumptions and models, to assure that fuel damage will not result from control rod malposition transients."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Function."

2. 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."

3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, pp. 241-248 (1967).

4. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 15.4.4
15.4.5

STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch

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

I. AREAS OF REVIEW

A number of transients that may occur with moderate frequency cause either increased core flow or introduction of cooler or de-borated water into the core. These transients result in an increase in core reactivity due to decreased moderator temperature, moderator boron concentration, or core void fraction. This review plan is intended to be applicable to all such transients. Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Ref. 1).

The specific transients (Table 15-1 of Ref. 1) evaluated are:

1. Boiling water reactor (BWR): startup of an idle recirculation pump.

2. BWR: flow controller malfunction causing increased recirculation flow.

3. Pressurized water reactor (PWR) with loop isolation valves; startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which the isolation valves open.

4. PWR without loop isolation valves: startup of a pump in an inactive loop.

The review of the core flow increase transients considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

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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.

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequences described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests CPB to initiate a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transients are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients described above are:

   a. To identify which of the transients are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel clad integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier, other than the fuel cladding, to the release of radioactive materials. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of the most limiting transients should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, they are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable:

   a. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty. An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two, or three loops initially operating) or the effects referenced to a limiting case.

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analyses, and the reviewer should compare these to the limiting safety settings included in the proposed technical specifications.

The description of the core flow increase transients presented in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that a particular core flow increase transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant should present a quantitative analysis in the SAR of the increase of flow transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of protection, engineered safety feature, and other systems needed to limit the consequences of the core flow increase transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7 and 8 of the SAR. The reviewer considers and evaluates the possibility of a single failure that would permit the loop isolation valves to open prior to startup of a pump in an idle loop (for those plants with loop isolation valves). If this could occur, the core flow rate increase would not be limited by the rate at which the valve opens, and the resulting rate of reactivity insertion could be greater than for other transients of this group.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that the selected core burnup yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. For each transient the variations with time during the transient of the core and barrier performance parameters listed in the "Event Evaluation" section of Chapter 15 of the Standard Format (Ref. 1) are reviewed. The values of the more important of these parameters for the core flow increase transients are compared to those predicted for other similar plants to see that they are within the range expected.

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 (SER):

"A number of plant transients can result in a core flow increase. Those that might be expected to occur with moderate frequency are the startup of an idle recirculation pump (BWR); flow controller malfunction causing increasing core flow (BWR); startup of a pump in an inactive reactor coolant loop (PWR); and startup of a pump in an initially isolated inactive reactor coolant loop.* All these postulated transients have been reviewed. It was found that the most limiting with regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient. This transient was evaluated by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (MDNBR)** did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam system pressures did not exceed 110% of the design values.

"The staff concludes that the plant design is acceptable in regard to transients that result in an increase in coolant flow through the reactor core."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41" (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

*The SER should present one statement for all similar transients.
**The minimum critical heat flux ratio or critical power ratio (MCHFR or MCPR) for a BWR.

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SECTION 15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

Unborated water can be added to the reactor coolant system, via the chemical volume and control system (CVCS), to increase core reactivity. This may happen inadvertently, because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator must stop this unplanned dilution before the shutdown margin is eliminated. Since the sequences of events that may occur depend on plant conditions at the time of the unplanned moderator dilution, the review includes conditions at the time of the unplanned dilution, such as refueling, startup, power operation (automatic control and manual modes), hot standby, and cold shutdown.

The review of postulated moderator dilution events considers causes, initiating events, the sequence of events, the analytical model, the values of parameters used in the analytical model, and predicted consequences of the event.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both the RSB and EISCSB. The RSB reviewer concentrates on the need for the reactor protection system and the operator action required to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

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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.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests CPB to initiate a generic evaluation of the new analytical model. APCSB reviews the functional and operational characteristics and potential failure modes of the CVCS, as described in Standard Review Plan (SRP) 9.3.4. The RSB reviewer makes use of this review to evaluate initiating causes and the expected sequence of events.

The predicted results of moderator dilution events are reviewed to assure that the consequences meet the acceptance criteria given in Section II of this SRP. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The general objective of the review of moderator dilution events is to confirm that either of the following conditions are met:

   a. The consequences of these events are less severe than the consequences of another transient that results in an uncontrolled increase in reactivity and has the same anticipated frequency classification.

   b. The plant responds to the events in such a way that the criteria regarding fuel damage and system pressure are met and adequate time is allowed the operator to terminate the dilution before the shutdown margin is eliminated.

2. The specific criteria for moderator dilution events are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 1).

   b. Fuel clad integrity should be maintained by ensuring that Section II, Acceptance Criterion 1 of SRP 4.4 is satisfied throughout the transient.

   c. Such incidents will not generate a more serious plant condition without other faults occurring independently.

   d. From the time an alarm makes the operator aware of unplanned moderator dilution, the following minimum time intervals must be available before a loss of shutdown margin occurs:

      (1) During refueling: 30 minutes.

      (2) During startup, cold shutdown, hot standby, and power operation: 15 minutes.

3. The applicant's analysis of moderator dilution events should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 2 through 4 are acceptable. Should other analytical methods be proposed, these methods must be evaluated by the staff. For new generic methods, the reviewer requests an evaluation by CPB.

All of the following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, and cold shutdown.

The parameters and assumptions used in the analytical model should be suitably conservative. The following values and assumptions are considered acceptable:

a. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2% to account for power measurement uncertainty.

b. The boron dilution is assumed to occur at the maximum possible rate.

c. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. This will usually be the beginning-of-life (BOL) condition.

d. All fuel assemblies are installed in the core.

e. A conservatively low value is assumed for the reactor coolant volume.

f. For analyses during refueling, all control rods are withdrawn from the core.

g. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1%) is assumed to exist prior to the initiation of boron dilution.

h. For each event analyzed, a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

i. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and set points used in the analysis will be preliminary in nature and subject to change. At the OL review, final values will be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The descriptions of moderator dilution transients presented in the SAR are reviewed by RSB regarding the occurrences leading to the initiating events. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to functior. Particularly important are the alarms which alert the operator to the unplanned boron dilution.

2. The extent to which the plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

The RSB reviewer confirms that analyses are included for a boron dilution incident occurring during each of the following plant initial conditions: refueling, startup, power operation (automatic control and manual modes), hot standby, and cold shutdown. For each such incident reviewed, all possible causes must have been considered by the applicant and justification presented that the cause selected for analysis is the one that allows the operator the least time to take corrective action.

With the aid of the EICSB reviewer, the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of each boron dilution incident to acceptable levels is reviewed. The RSB reviewer compares the predicted variations of system parameters with various trip and system initiation set points. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation where the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components that may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, 8, and 9 of the SAR. In particular, the redundancy of alarms that alert the operator to the unplanned dilution is confirmed.

The mathematical models used by the applicant to evaluate core performance and reactivity status are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant. The justification provided by the applicant to show that the selected core burnup condition, boron concentration, and rod worths yield the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity

parameters used in the applicant's analysis. These values are reviewed by CPB under SRP 4.2. The value of core reactivity as a function of time following each incident analyzed is confirmed by comparison with an acceptable analysis performed for another plant, by comparison with staff calculations for typical plants done by CPB on request, or by independent calculations by the RSB reviewer.

The assumed dilution flow rates are reviewed, taking into consideration the system parameters which act to limit the flow. The reviewer examines the flow-limiting equipment characteristics provided by the applicant to justify his flow rate assumptions; e.g., if the flow is limited by the charging pump capacity, the assumed flow is compared with the flow for all charging pumps acting at full capacity. If some lesser value of flow is assumed, such as not all pumps operating, or flow limited by a valve, justification must be provided. EICSB is consulted concerning any interlocks for which credit is taken.

The results of the analyses are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the time available for the operator to take corrective action. The variations with time during the transient of important parameters are compared to those predicted for other similar plants to see that they are within the range expected. Parameters of particular importance are core reactivity, boron concentration, rate of addition of unborated water, power level, core pressure, and minimum departure from nucleate boiling ratio (DNBR).

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:

> "Various chemical and volume control system (CVCS) malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The malfunctions that allow the operator the shortest time for corrective action have been analyzed starting from plant conditions of startup, power operation (automatic and manual), hot standby, cold shutdown, and refueling. These events were evaluated by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analyses of these events showed that the operator has ___ minutes to take corrective action if a boron dilution incident occurs during refueling and ___ minutes if at power. In the latter case,the most severe transient results in a minimum departure from nucleate boiling ratio (DNBR) of ____ and reactor coolant and main steam system pressures of less than 110% of design.

"Based on these results, the staff concludes that the plant design is acceptable with regard to boron dilution incidents caused by malfunctions of the chemical and volume control system."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

2. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41" (under review).

3. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

4. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

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

6. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
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SECTION 15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Accident Analysis Branch (AAB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The review of fuel loading errors considers:

1. The spectrum of misloading events analyzed. A sufficient number of fuel loading errors must be studied by the applicant and presented to show that the worst situation undetectable by incore instrumentation has been identified. The kinds of errors considered should include loading of one or more fuel assemblies into improper locations and, where physically possible, with incorrect orientation. For those reactors in which burnable poison or fuel rods are added to or removed from assemblies at the plant, errors in these processes must be considered.

2. Changes in the power distribution and increased local power density.

3. The provisions made to search for loading errors at the beginning of each fuel cycle.

CPB also reviews the effect of misloaded fuel on nuclear design parameters, the detection of fuel loading errors, and any operational restrictions that would assist in staying within safety limits.

AAB reviews the radiological implications of misloaded fuel on request.

On request, QAB reviews the measures provided to minimize the probability of fuel misloading.

II. ACCEPTANCE CRITERIA

The acceptance criteria are based on General Design Criterion 13 (Ref. 1) and on 10 CFR Part 100 (Ref. 2). Criterion 13 is concerned with instrumentation and control. One intent of this criterion is to require monitoring of variables over anticipated

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ranges for normal operating and accident conditions. The consequences of misloaded fuel that cannot be detected are covered by 10 CFR Part 100, which is concerned with reactor site criteria.

The primary safeguards against fuel loading errors are procedures and design features to minimize the likelihood of the event. Additional safeguards include incore instrumentation systems which would detect errors. However, should an error be made and go undetected, it is possible in some reactor designs for fuel safety limits to be exceeded. Therefore, additional acceptance criteria are required to cover the event of operation with misloaded fuel.

The acceptance criteria for fuel loading errors are:

1. The plant technical specifications should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

2. In the event the error is not detectable by the instrumentation system and fuel safety limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

III. REVIEW PROCEDURES

The review procedures for fuel loading errors are as follows:

1. The reviewer verifies that the various cases of misloaded fuel assemblies outlined in Section I have been analyzed by the applicant and the worst case determined. For each case the effect on the reactor power distribution should be given.

2. The reviewer determines that the effect each postulated error has on reactor instrumentation has been ascertained. For limiting events (where safety limits are exceeded) the reviewer verifies that acceptable techniques (Ref. 4) have been used to calculate the fuel temperature conditions.

3. The reviewer assures compliance with Criterion 1 of Section II above by reviewing the plant technical specifications to verify that they contain provisions requiring that incore instrumentation be used to search for misloaded fuel after each fueling operation. Since low-power mapping is typically done, searching for misloading can be accomplished by the usual low-power maps.

4. When it is determined that fuel limits can be exceeded, AAB is requested to perform dose calculations to assure that Criterion 2 of Section II above is met.

IV. EVALUATION FINDINGS

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

"The staff has evaluated the consequences of a spectrum of postulated fuel loading errors. We conclude that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable but the offsite consequences of any core damage are a small fraction of 10 CFR Part 100 guidelines. The applicant affirms that the available incore instrumentation will be used before the start of a fuel cycle to search for fuel loading errors."

V. REFERENCES

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

2. 10 CFR Part 100, "Reactor Site Criteria."

3. Standard Review Plan 4.2, "Fuel System Design."

4. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

U.S. NUCLEAR REGULATORY COMMISSION
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SECTION 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

The CPB evaluates the consequences of a control rod ejection accident in the area of physics. The review covers the possible initial conditions, rod patterns and worths, scram worth as a function of time, adequacy of the various reactivity coefficients, adequacy of the calculational methods, and any core parameters which affect the peak reactor pressure or the probability of fuel pin failure.

The AB reviews the relevant thermal-hydraulic analyses under SRP Section 4.4.

The AAB reviews, as described in the appendix to this SRP section, the radiological consequences of a rod ejection accident by using a source term for dose calculations based on the amount of failed fuel as obtained by CPB from the physics and thermal-hydraulic analyses.

ICSB in SRP Sections 7.2 and 7.3 reviews the applicant's determination of the reactor trip delay time, i.e., the time elapsed between the instant the sensed parameter reaches the level for which protective action is required and the onset of negative reactivity insertion.

II. ACCEPTANCE CRITERIA

Regulatory Guide 1.77 (Ref. 1) identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident. Two criteria are used by CPB in evaluating the rod ejection accident:

1. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed transient should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code (Ref. 2).

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The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling (DNB) condition is an input to the radiological evaluation by AAB. The radiological criteria used in the evaluation of rod ejection accidents (PWRs) are given in Appendix B of Regulatory Guide 1.77 (Ref. 1).

III. REVIEW PROCEDURES

1. Review of the applicant's analyses, showing that the first of the acceptance criteria above is met, proceeds as follows:

   a. A spectrum of initial conditions is considered, which must include both zero-power and full-power conditions, at beginning and end of fuel lifetime (BOL and EOL), to assure examination of upper bounds on possible fuel damage. Initial full-power conditions should include the uncertainties in the calorimetric measurement of power.

   b. From the initial conditions of (a) and from control rod patterns, the limiting rod worth is determined. Where confirmation is considered necessary the reviewer may calculate, as an audit, the worth of limiting rods.

   c. Reactivity coefficient values corresponding to the limiting initial conditions must be used at the beginning of the transient. The reviewer checks the reactivity coefficient curves used by the applicant with those reviewed by the CPB under SRP Section 4.3. The two coefficients of most interest are the Doppler and moderator coefficients. If no three-dimensional space-time calculation is performed, the reactivity feedback must be conservatively weighted to account for the variation in the missing dimension(s).

   d. The reviewer inspects the control rod insertion assumptions which include: trip parameters, trip delay time, rod velocity curve, and differential rod worth. Trip parameters and delay time are covered under SRP Section 7.2 by ICSB. Rod worth is checked by the reviewer for consistency with the review performed under SRP Section 4.3.

   e. The applicant's analytical methods are reviewed. The reviewer may use the results of previous case work, if the analytical methods have been previously reviewed and approved by the staff. Otherwise he must perform a complete review on this case. Alternatively an audit of several calculations, using methods considered acceptable to the staff, may be done by the reviewer (or consultants to the staff). The primary concern of the reviewer is how well the analytical model elements represent the true three-dimensional problem. Other items checked by the reviewer include feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

f. Results of the calculations done by procedures described in steps a-e are expressed as values of the radially-averaged fuel rod enthalpy (in units of cal/gm). The reviewer determines that the maximum value does not exceed 280 cal/gm.

2. Verification of compliance with the second acceptance criterion is accomplished as follows:

   a. The same procedures considered in steps a-f above are followed.

   b. For each accident, the transient primary system pressure should be calculated by an analytical method acceptable to the staff or, as before, an independent audit calculation is made by the staff. The reviewer checks the results (as obtained by the applicant or the staff) for compliance with the second criterion.

3. The number of fuel rods experiencing clad failure is determined (for use in evaluating the radiological consequences) by the following procedure:

   a. The reviewer determines that an acceptable procedure for calculating a departure from nucleate boiling condition during the reactivity excursion has been used. This may be done by referring to previous cases for the same nuclear steam supply system (NSSS) vendor. If no approved technique is available, as might be the case for the first project using a new or substantially revised model, the reviewer must perform a separate detailed review (which may be documented separately in a topical report).

   b. The reviewer must determine that the number of rods used in the radiological evaluation is the number of rods calculated to have a departure from nucleate boiling. Departure from nucleate boiling must be calculated in accordance with the criteria reviewed and accepted by the AE under SRP Section 4.4. Typically, the criterion defines a departure from nucleate boiling ratio (DNBR) less than 1.30 when DNB correlations such as W-3 (Ref. 3) or BAW-2 (Ref. 4) are used.

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 staff has evaluated the applicant's analysis of the assumed rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore,

calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

4. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)

APPENDIX

STANDARD REVIEW PLAN 15.4.8

RADIOLOGICAL CONSEQUENCES OF
CONTROL ROD EJECTION ACCIDENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Core Performance Branch (CPB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The AAB review under this appendix covers the following areas:

1. The plant response to a control rod ejection accident.

2. The calculation of whole body and thyroid doses at the exclusion area boundary and low population zone outer boundary due to the releases resulting from a rod ejection accident.

The purposes of the review are to assure that the plant procedures for recovery from a rod ejection accident and the plant technical specifications are properly taken into account in computing the whole body and thyroid doses at the nearest exclusion area boundary and low population zone (LPZ) outer boundary, and to compare the calculated doses against the appropriate guidelines.

The physics and thermal-hydraulic aspects of the accident are reviewed by CPB. Verification of the applicant's calculations of the number of fuel pins experiencing departure from nucleate boiling (DNB) and the amount of fuel reaching the clad melting temperature is obtained from the CPB.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a rod ejection accident at the construction permit stage, and the primary-secondary system leakage appropriately limited, if calculations show that the resulting doses at the nearest exclusion area boundary are on the order of 150 rem to the thyroid and 20 rem to the whole body, or less, for the first two hours after the accident, and 150 rem thyroid and 20 rem whole body, or less, for the course of accident at the LPZ outer boundary. Higher doses may be acceptable at the operating license (OL) review stage, up to the guidelines of 10 CFR Part 100. Technical specifications should be set to assure that the doses resulting from a rod ejection accident are limited to the guideline values.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix as may be appropriate for a particular case. The judgment on areas 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 detailed review of the radiological consequences of a rod ejection accident is done at OL stage when system parameters and accident analysis results are fully developed. At the CP stage, the reviewer estimates the doses from the rod ejection accident based on results from similar plants that have been recently reviewed.

The AAB review of the rod ejection accident at the OL stage covers the following topics:

1. Release of the radioisotopes to the environment via the containment building.

2. Release of radioisotopes to the environment through the secondary system.

3. Calculation of resulting doses.

Physical plant parameters, such as the steam generator steaming rates, are reviewed to ascertain their conservatism.

Regulatory Guide 1.77 (Ref. 2) should be used as a guide in the analysis of the accident. The release of radioisotopes through the secondary system should be analyzed independently by means of a digital computer code. Computer codes are currently under development within NRC. Documentation will be published in a NUREG report. In the analysis of this accident, a loss of offsite power is assumed. It is also assumed that nuclides released to the primary coolant due to any fuel failures or melting (this information is obtained from the CPB) are instantaneously and uniformly mixed in the coolant at the time of the accident. For releases via the containment building, Regulatory Guide 1.77 recommends that 100% of the noble gases and 25% of the iodines contained in the fuel which is estimated to reach initiation of melting be available for release from the containment. For releases through the secondary system, 100% of the noble gases and 50% of the iodines contained in the fuel which is estimated to reach initiation of melting are assumed to be released to the primary coolant.

The SAB provides the reviewer with the distance to the nearest boundary of the exclusion area, the accident (5 percentile) wind speed and X/Q, and the 0-8 hr and 8-24 hr X/Q values at the outer boundary of the LPZ. These X/Q values are used to estimate the consequences of releases from the containment and the consequences at the LPZ outer boundary of releases from the secondary system. The X/Q value for calculating the two-hour dose at the nearest exclusion area boundary from the releases through the secondary system is obtained from Regulatory Guide 1.5 (Ref. 3) and corrected for wind speeds differing from 1 m/sec (inverse ratio). A breathing rate of 3.47×10^{-4} m^3/sec is used in calculating the thyroid doses for the first 8 hours after the accident; from 8 to 24 hours, a breathing rate of 1.75×10^{-4} m^3/sec is used.

Although the resulting doses in case of an actual accident would be a composite of the doses computed for releases via the containment building and through the secondary system, both doses should be presented. If either dose approaches the limit, calculation of representative composite cases should be considered (the AAB branch chief should be consulted).

If the doses resulting from the releases through the secondary system exceed the limits specified in Section II above, the technical specification limit on primary-secondary system leakage is reduced accordingly. If the doses resulting from the potential releases from the primary containment exceed the specified limits, the pressure setpoint for actuation of the containment sprays may have to be reduced to obtain credit for spray removal of the fission products.

The physics and thermal-hydraulic aspects of the accident are reviewed by the CPB. Verification of the applicant's calculations of the number of fuel pins reaching DNB and the amount of fuel reaching the fuel melting temperature are obtained (and documented by buckslip) from the CPB. It is important to note that the fuel melting temperature criterion used for release of large fractions of fission gases corresponds to the initiation of melting as opposed to the 280 cal/gm used as a criterion by the CPB for core disruption.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions such as the following, to be included with the CPB findings in the staff's safety evaluation report at the operating license stage:

> "The consequences of the rod ejection accident have been evaluated, and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the recommendations of Regulatory Guide 1.77, the computer code _____, and a conservative description of the plant response to the accident. The calculated doses are presented in Table _____. Technical specification limits on primary-secondary coolant leakage assure that the potential doses are well within 10 CFR Part 100 exposure guidelines."

At the construction permit stage the following paragraph is included with the CPB findings in the staff's safety evaluation report:

> "On the basis of our experience with the evaluation of the control rod ejection accident for PWR plants, we have estiamted the doses from this accident to be on the order of _____rem to the thyroid and _____rem to the whole body at the nearest boundary of the exclusion area. If a reevaluation of this accident at the operating license stage results in dose estimates that exceed the 10 CFR Part 100 guidelines, appropriate limits on primary-secondary leakage and the setpoint for containment spray actuation will be set."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

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

3. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."

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SECTION 15.4.9 SPECTRUM OF ROD DROP ACCIDENTS (BWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

CPB evaluates the consequences of a rod drop accident in a boiling water reactor (BWR) in the area of physics. The CPB review covers the applicant's description of the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, and methods used to analyze the accident. A general reference on rod drop accident analysis is noted in Reference 1.

The Analysis Branch, under SRP Section 4.4, reviews the thermal and hydraulic analyses.

AAB, as described in the appendix to this SRP section, reviews the radiological consequences of a rod drop accident, using the amount of failed fuel as obtained by CPB from the physics and thermal-hydraulics analyses as the source for dose calculations.

ICSB, in SRP Sections 7.2 and 7.3, reviews the reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion.

II. ACCEPTANCE CRITERIA

CPB focuses on two limiting criteria in evaluating the rod drop accidents:

1. Reactivity excursions should not result in radially averaged fuel rod enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed transient should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code (Ref. 2).

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The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

III. REVIEW PROCEDURES

1. Review of the applicant's analyses showing compliance with the first of the above criteria is carried out as follows:

   a. The reviewer verifies that the applicant has considered a spectrum of initial conditions for this event that covers the range of time-in-cycle and initial power levels. Initial full power conditions should include the uncertainties in the calorimetric measurement of power.

   b. The reviewer verifies that the maximum expected individual rod worths are used. In developing rod worth criteria, the nominal control rod withdrawal pattern must be considered, as well as those abnormal patterns that are not precluded by an instrumentation system acceptable to ICSB.

   c. The reviewer determines that an acceptable and conservative function is used to describe the control rod worth as a function of control rod position and that the control rod position as a function of time is suitably conservative.

   d. The reviewer determines that conservative reactivity coefficients, notably the Doppler coefficient, are used and that they are compatible with those described in SRP Section 4.3.

   e. The reviewer assures that the scram action is conservatively represented in the use of the integral scram worth curve (SRP Section 4.3) and in the use of the scram delay time.

   f. The reviewer checks the analytical methods or assures that they have been reviewed and approved previously. The reviewer may also perform an independent audit calculation using methods acceptable to the staff. The applicant's methods should account conservatively for all major reactivity feedback mechanisms.

2. The reviewer inspects the results of the calculation of maximum reactor pressure to determine compliance with the second criterion listed in subsection II (the reviewer may do an audit calculation when appropriate).

3. The number of fuel rods experiencing clad failure is determined (for use in evaluating the radiological consequences) by the following procedures:

   a. The reviewer determines that the transient critical power ratio (CPR) has been computed by an acceptable technique (either previously reviewed or reviewed de novo during this review).

b. The reviewer determines that the number of rods with enthalpy exceeding 170 cal/gm has been computed by an acceptable method.

c. The reviewer determines that the amount of fuel exceeding melting conditions has been computed by an acceptable method.

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 staff has evaluated the applicant's analysis of the assumed rod drop accident and finds the assumptions, calculational techniques, and consequences acceptable. Since the calculations predict peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molted UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III of the ASME Boiler and Pressure Vessel Code) for the maximum rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained."

V. REFERENCES

1. "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, General Electric Company, March 1972; Supplement 1 to NEDO-10527, July 1972; and Supplement 2 to NEDO-10527, January 1973.

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

APPENDIX

STANDARD REVIEW PLAN SECTION 15.4.9

RADIOLOGICAL CONSEQUENCES OF
CONTROL ROD DROP ACCIDENT (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Core Performance Branch (CPB)
Hydrology-Meteorology Branch (HMB)

I. AREAS OF REVIEW

The AAB review of the postulated control rod drop accident covers the release of noble gases and iodine fission products from the core to the environment via the turbine and condensers and a determination of the doses as a result of the accident. The areas reviewed under this appendix include an examination of the plant responses to this accident and an evaluation of the radiological consequences including the thyroid and whole body doses at the exclusion area and low population zone (LPZ) outer radius boundaries to assure that the consequences meet the acceptance criterion given in Section II below.

The core response aspects of the accident are reviewed by the CPB. Verification of the applicant's calculation of the number of fuel rod failures and the amount of fuel reaching the melting temperature is obtained from the CPB.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a control rod drop accident and the consequences acceptable if the resulting doses at the exclusion area and LPZ outer boundaries are well within (taken to be less than or equal to 25%) the guideline values of 10 CFR Part 100 (Ref. 1).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix 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.

Based on past reviews, a control rod drop accident is expected to result in radiological consequences which are less than 10% of the Part 100 guideline values even with very conservative assumptions. The reviewer should examine the site meteorology, plant features, and fuel damage as a result of the accident for the plant in question and compare these with the corresponding features and resulting doses for previously reviewed plants to ascertain whether a specific calculation of the radiological consequences should be performed. Unless unusual plant or site features are present or the applicant's calculation

shows an unusually large amount of fuel damage, a specific calculation of the radiological consequences is not necessary and a comparison of the pertinent plant and site features is sufficient to conclude that the consequences of this event meet the acceptance criterion given in Section II. However, a specific evaluation of this accident should be performed for the first application involving a particular standardized design to establish a reference point for comparison of future applications incorporating the design.

Where a specific calculation of the radiological consequences is to be performed, the procedure is as follows:

The core response aspects of the accident are reviewed by the CPB. Verification of the applicant's calculation of the number of fuel rod failures and the amount of fuel reaching the fuel melting temperature is obtained from the CPB.

The following assumptions regarding the plant condition and release and transport of radioactivity are used:

1. A coincident loss of offsite power is assumed at the time of the accident.

2. The integrity of the turbine and condensers is unaffected by the rod drop accident.

3. The combination of reactor operating mode, control rod positions, core burnup, etc., that results in the largest source term is selected for evaluation.

4. No allowance is made for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

5. The amount of activity accumulated in the fuel-clad gap is assumed to be the same as that in Regulatory Guide 1.77.

6. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842°C) at any time during the course of the accident is calculated and 100% of the noble gases and 50% of the iodines contained in this fraction are assumed released to the reactor coolant. CPB should be requested to review analyses which propose that fuel melting is not likely to result in significant releases prior to MSIV closure.

7. Those fuel rods presumed to fail are assumed to have operated at power levels 1.5 times that of the average power level of the core.

8. Any nuclides released to the reactor coolant from fuel cladding failures or fuel melting are instantaneously and uniformly mixed in the reactor coolant in the pressure vessel at the time of the accident.

9. A conservative analysis may assume that 10% of the iodines and 100% of the noble gases released in the pressure vessel reach the turbine and condensers. A more realistic analysis which accounts for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation valve (MSIV) and which considers the MSIV closure time may be performed as needed on a case-by-case basis.

10. All noble gases remain in a gaseous state and are available for leakage from the turbine and condensers.

11. Of those iodines which reach the turbine and condensers, 90% are removed by partitioning and plateout in the turbine and condensers leaving 10% airborne and available for leakage.

12. The turbine and condensers leak to the atmosphere at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. Condenser leakage rates lower than 1% per day and shorter in duration than 24 hours will be reviewed on a case-by-case basis. Credit for condenser vacuum discharge isolation on high activity level in the steam, or credit for filtration of the condenser vacuum discharge, will also be reviewed on a case-by-case basis.

13. The effects of radiological decay during holdup in the turbine and condensers are taken into account.

14. The X/Q values (obtained from the HMB), breathing rates and dose conversion factors are the same as those used in the calculation of doses from a loss-of-coolant accident (Ref. 2).

The above assumptions are used in conjunction with a branch-approved computer code such as TACT to compute the radiological consequences.

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 with the CPB findings in the staff's safety evaluation report:

Where the radiological consequences have not been specifically calculated, the findings may be in the following form:

> "Although the radiological consequences of a postulated rod drop accident have not been specifically calculated for the _____ plant, we conclude, based upon a review of this accident for similar plants at a number of sites, and based upon the similarity of those plant features for the _____ plant which affect the radiological consequences, that the consequences for this event are well within the guideline values of 10 CFR Part 100 and are acceptable."

Where the radiological consequences have been calculated, the findings may be of the following form:

"The consequences of the rod drop accident have been evaluated and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using an analytical model based upon a conservative description of the plant response to the accidents. The calculated doses are presented in Table _____, and are well within the 10 CFR Part 100 exposure guidelines."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Appendix A, SRP Section 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident (Containment Leakage Contribution)."

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SECTION 15.5.1 INADVERTENT OPERATION OF ECCS AND CHEMICAL AND VOLUME
15.5.2 CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR
COOLANT INVENTORY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

Various types of equipment malfunctions, operator errors, and abnormal occurrences that may occur with moderate frequency can cause an unplanned increase in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and lead to fuel damage or overpressurization of the reactor coolant system. Alternatively, a power level decrease and depressurization may result. In either case, if the transient is severe enough the reactor will trip from high water level, high flux, or high or low pressure. For pressurized water reactors (PWR's) operator action will probably be required to terminate the unwanted injection flow.

This review plan is intended to be applicable to these types of moderate frequency events that increase reactor coolant inventory. These transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Ref.1). The specific initiating events considered in this review plan are:

1. Boiling water reactors (BWR's) - Inadvertent operation of the high pressure core spray, high pressure coolant injection, or reactor core isolation cooling system.

2. PWR's - Inadvertent operation of high pressure emergency core cooling system (high pressure injection system) or a malfunction of the chemical and volume control system.

Other BWR transients that can result in an increase in reactor coolant inventory include feedwater system malfunctions (increasing flow), steam pressure regulator malfunctions (decreasing flow), loss of electrical load, turbine trip, main steam isolation valve (MSIV) closure, and loss of condenser vacuum. These transients are the subject of other standard review plans that consider their effects on system parameters other than coolant

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inventory. However, the impact of these transients on reactor coolant inventory is considered by the reviewer as a portion of the effort involved in this review plan.

The review of events leading to an increase in reactor coolant inventory considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of those transients analyzed are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are:

   a. To identify which of the moderate frequency* events leading to a coolant inventory increase are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds to the core flow increase in such a way that the criteria regarding fuel damage and system pressure are met.

*The term "moderate frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 8 and 9.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel clad integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 (Ref. 7) is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model are to be suitably conservative. The use of the following values is considered acceptable:

   a. The reactor is initially at rated power (licensed core thermal power) plus 2% for power measurement uncertainty.

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicant's description of events leading to an increase in reactor coolant inventory is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

The applicant should present a quantitative analysis in the SAR of the event leading to an increase in reactor coolant inventory which is the most limiting. For this event, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the event to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. In this phase of the review, the system reviews are performed as described in the standard review plans for Chapters 4, 5, 6, 7, 8, and 9 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in Section II regarding maximum pressure in the reactor coolant and main steam systems and the minimum critical heat flux ratio (MCHFR) or departure from nucleate boiling ratio (DNBR). The variation with time during the transient of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The values of the more important of these parameters for the events leading to an increase in reactor coolant inventory are compared to those predicted for other similar plants to confirm that they are within the expected range.

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 (SER):

"A number of plant transients can result in an increase in reactor coolant inventory. Those that might be expected to occur with moderate frequency are inadvertent operation of the emergency core cooling system, chemical and volume control system malfunction, and various BWR transients.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the _____ transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)** did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with regard to events that are expected to occur with moderate frequency and to result in an increase in reactor coolant inventory."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

*The SER draft should present one statement for all similar transients.
**Minimum critical heat flux ratio for a BWR.

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SECTION 15.6.1 INADVERTENT OPENING OF A PWR PRESSURIZER SAFETY/RELIEF VALVE OR A BWR SAFETY/RELIEF VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

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

I. AREAS OF REVIEW

The inadvertent opening of a safety or relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. The effect of the pressure decrease is to decrease the neutron flux (via moderator density feedback). In a pressurized water reactor (PWR), a reactor trip occurs due to low reactor coolant system (RCS) pressure. In a boiling water reactor (BWR), the safety or relief valve discharges into the suppression pool. Normally there is no scram in a BWR. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCV) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell.

The review of these transients should consider the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

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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.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient are reviewed to assure that the consequences meet the acceptance criteria given in Section II of this standard review plan (SRP). Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The general objective in the review of inadvertent primary safety or relief valve opening events is to confirm that one of the following criteria is met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of reactor coolant inventory and has the same anticipated frequency classification.

   b. The plant responds to the safety or relief valve opening transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 1).

   b. Fuel clad integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 (Ref. 7) is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of this transient should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 2 through 5 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model are to be suitably conservative. The use of the following values is considered acceptable:

a. The reactor is initially at 102% of licensed core thermal power (2% allowance for power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicant's description of the inadvertent safety or relief valve opening transient is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that the inadvertent safety or relief valve opening transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the transient is presented in the SAR, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may alter the course of the transient. In this phase of the review, the system reviews are performed as described in the standard review plans for Chapters 5, 6, 7 and 8 of the SAR. The reviewer considers possible single failures in systems that replenish or maintain the reactor coolant inventory.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II regarding the maximum pressure in the reactor coolant and main steam systems. The variation with time during the transient of the core and barrier performance parameters listed in the Event Evaluation Section of Chapter 15 of the Standard Format (Ref. 6) are reviewed. Values of the more important of these parameters for the transient caused by inadvertent safety or relief valve opening are compared to those predicted for other similar plants to confirm that they are within the expected range.

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 (SER):

> "A number of plant transients can result in a decrease of reactor coolant inventory. Those that might be expected to occur with moderate frequency are safety or relief valve openings, minor primary pipe breaks, and (in BWR's) loss of feedwater.* All of these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the _____ transient showed that cladding integrity was maintained by ensuring that the minimum departure from

*The SER draft should present one statement for all similar transients.

nucleate boiling ratio (DNBR)* did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable in regard to transients that are expected to occur with moderate frequency and result in a decreased primary coolant inventory."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

2. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

3. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41" (under review).

4. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

5. "Standard Nuclear Steam System, B-SAR-241," Babcock and Wilcox Company, February 1974 (under review).

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

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

*Minimum critical heat flux ratio or critical power ratio (MCHFR or MCPR) for a BWR.

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SECTION 15.6.2

RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Hydrology-Meteorology Branch (HMB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

This SRP section covers the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary, such as instrument lines and sample lines. For the purpose of this review these lines are classified into two categories:

1. Those lines meeting the isolation requirements of General Design Criterion 55.

2. Lines exempted from General Design Criterion (GDC) 55, i.e., instrument lines.

Specifically, the AAB review covers the following areas for lines exempt from the GDC 55 isolation requirements:

1. The calculation of whole body and thyroid doses at the site boundary resulting from a failure of an instrument line.

2. Any limitations on primary coolant radioactivity concentrations, as established in the technical specifications, required as a result of the analysis of instrument line failures.

For lines which meet the General Design Criteria 55 isolation requirements the AAB reviews the requirements, if any, on the isolation time and maximum leak rate of the isolation valves, which may be necessary to assure that the radiological consequences of a failure of these lines will not exceed a small fraction of 10 CFR Part 100 dose guidelines.

The RSB may be requested by the AAB to confirm the value assumed for the mass of coolant released in the accident and to determine if this accident will cause fuel failures. The HMB provides the reviewer with the X/Q values to be used in the two-hour dose

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calculations. The CSB should be requested by the reviewer to verify that secondary containment integrity and leaktightness are maintained during the course of the accident.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against the radiological consequences of a failure of containment-penetrating small line carrying reactor coolant (and the technical specifications for primary coolant activity and isolation time and maximum allowable leak rate of isolation valves in these lines appropriately limited) if calculations show that the resulting doses at the exclusion area boundary are small fractions (no more than 10%) of the 10 CFR Part 100 exposure guidelines.

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

This event is an infrequent incident for which a scoping analysis was performed by the staff on Dresden Unit 2 in October of 1975. This analysis indicated that significant doses at the exclusion area boundary could result if safety measures were not present to restrict the primary system fluid loss from a postulated small line failure. To protect against the possibility of significant doses, 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.

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 AAB review at the CP and OL stage consists of the following steps:

1. Review of the applicant's descriptions of the accidents, in order to confirm the appropriateness and conservatism of the assumptions used in the analysis.

2. Performance of an independent analysis of the radiological consequences of the failure of the instrument or other line. The following conservative assumptions are used to simplify the analysis:

    a. All lines exempt from General Design Criteria 55 meet the requirements of Regulatory Guide 1.11 and its supplement. Specifically, each line must be sized or include flow restrictions, such as 1/4" orifices within the primary containment, to limit the flow in the event of a line rupture.

In addition, a single isolation valve capable of automatic operation or a self-actuated excess flow check valve capable of performing the same safety functions as an isolation valve should be located outside of and as close as practical to the primary containment. The status of this valve must be indicated in the control room at all times to assure its effectiveness in the event of failure of a line. In our analysis, we assume that an instrument line breaks at the primary containment wall on the secondary side upstream from the outboard isolation valve. It is further assumed that an iodine spike occurs and all of the iodine in the coolant which flashes to steam becomes airborne and available for release to the atmosphere.

b. All piping beyond the primary containment wall up to an including the outboard isolation valve or excess flow check valve must be seismic Category I.

c. The mass of reactor coolant released during a two-hour period is estimated with the assumption of choked flow at a fluid enthalpy equal to that of the reactor coolant under normal operating conditions. (Assume this is correct unless otherwise notified by RSB.)

d. The initial fission product concentrations in the primary coolant are those given as maximum equilibrium values in the technical specifications. The effects of an activity spike resulting from reactor shutdown or depressurization of the primary system is modeled by increasing the activity release rate from the fuel by a factor of 500.

e. The fission products in the primary coolant are assumed to be released to the environment if the line carrying the primary coolant penetrates the secondary containment (if any). Otherwise, the fission products in the coolant which flash to steam are assumed to be released to the secondary containment or auxiliary building, as appropriate. The CSB should verify the integrity and leaktightness of the secondary containment during the pressure transient associated with a failure within its boundary. An appropriate mixing volume is determined from the location of the assumed failure and the proximity to secondary containment ventilation systems assumed to be operating (if any).

f. Depending on the operability of any air treatment system(s), a ground-level or elevated (stack) release is assumed. The HMB provides the reviewer with the X/Q value.

g. Upon request by the reviewer, the RSB will determine the extent of damage to the fuel, if any, as a result of the accident and provide this information to the AAB.

3. Review of the results of the dose calculations. The resulting doses from these accidents should be small fractions of the 10 CFR Part 100 limits. If this is not the case, the primary coolant activity concentration limits allowed by the technical specifications should be reduced.

4. A survey of all other lines carrying primary coolant and penetrating the containment to determine the need for any special technical specification limits for the isolation valves in these lines. A failure of the line downstream of the outboard isolation valve, combined with a single active failure of any valve in the line is postulated. If such an accident sequence could result in significant primary coolant release, it may be necessary to specify a maximum allowable valve closure time and maximum allowable leak rate for the isolation valves in the technical specifications to assure that the consequences of this accident sequence do not exceed small fractions of the exposure guidelines of 10 CFR Part 100.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report at the construction permit and operating license stages:

"The applicant's analyses of the failure of small lines carrying primary coolant outside the containment and the proposed technical specifications for limiting the consequences of such failures have been reviewed. Independent evaluations of these accidents have been performed, and we conclude that the technical specifications for the reactor coolant activity and the isolation times and leakage limits for the isolation valves in such lines [if appropriate] assure that the radiological consequences of these accidents are limited to small fractions of the dose guidelines of 10 CFR Part 100."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Containment," and its Supplement, "Backfitting Considerations."

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SECTION 15.6.3 RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Hydrology and Meteorology Branch (HMB)
Reactor Systems Branch (RSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The purposes of the review are to assure that the plant procedures for recovery from a steam generator tube failure, with and without offsite power available, are properly taken into account in the computation of whole body and thyroid doses at the exclusion area and low population zone boundaries, and to assure that releases from the failure are adequately limited by the coolant activity concentration technical specifications. The RSB will notify the AAB if this accident is predicted to cause fuel failures, with and without a control rod held in the fully withdrawn position. The HMB will provide the reviewer with appropriate X/Q values for this event. The ETSB provides the AAB with appropriate assumptions and models regarding iodine spiking and its effect upon coolant activity.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a steam generator tube failure, and the primary and secondary coolant activities adequately limited, if calculations show that the resulting doses at the exclusion area and low population zone boundaries are less than a small fraction (less than 10%) of the 10 CFR Part 100 exposure guidelines, and are within 10 CFR Part 100 guidelines for the case of a preaccident iodine spike or for the case of one rod held out of the core.

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 during 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.

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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.

At the construction permit (CP) stage, the review is limited to a brief survey of the pertinent portions of the plant design and the applicant's discussion of this accident to determine that there are no unusual features that would prevent limitation of radiological consequences to acceptable levels by appropriate limits on coolant activity concentrations. The detailed review of the radiological consequences of a steam generator tube failure is done at the operating license (OL) stage when system parameters and accident analyses are fully developed. Standard Technical Specifications for each of the three PWR vendors' NSSS include limits on the primary and secondary coolant activities and primary-to-secondary leak rate. If the applicant proposes to use these standard limits and his plant is one of the standard NSSS/BOP plants for which the tube failure accident has been evaluated generically with the standard coolant activity and leakage limits, the reviewer need not reevaluate the offsite doses from this accident if the X/Q's for the site under review are lower than the limiting X/Q used in the generic review of the standard plant tube failure accident.

The AAB review of the steam generator tube failure accident at the OL stage consists of the following:

1. Review of the applicant's description of the tube failure accident (with and without offsite power). This includes a review of the sequence of events in the description, the bases for their occurrence, and assurance of an adequate degree of conservatism.

2. Review of the signals available to the reactor operator that indicate the occurrence of the accident and the state of the system throughout the recovery procedure. Automatic and required manual operations by the operator as a function of time must also be determined. The RSB should be contacted for a determination of the acceptability of the applicant's description of events, including operator actions.

3. Post-accident thermohydraulic characteristics and radiological consequences of this accident are highly plant-specific. The reviewer, with the use of a branch-approved code, computes post-accident thermohydraulic profiles and compares these with those presented by the applicant. The purpose of such comparison is not to attain exact match but to confirm the validity of the applicant's calculated results.

4. Meteorology. The AAB provides the HMB with the release conditions and location. The HMB then provides the AAB with the appropriate EAB and LPZ X/Q values for this event.

5. Determination of the initial coolant activity concentrations. The reviewer assumes the primary and secondary coolant activity concentrations allowed by the technical specifications (SAR Chapter 16 or the Standard Technical Specifications given in References 5, 6 and 7) as equilibrium conditions prior to the accident.

6. Determination of the iodine spiking effects. Two cases are analyzed, one with an iodine spike assumed to begin because of the reactor trip or primary system depressurization when the tube failure occurs, and one with an iodine spike assumed to have begun well before the accident due to a previous reactor transient (Refs. 3, 4).

The ETSB provides the AAB with the iodine spiking model. If the applicant proposes to use a spiking model different than the model described below, the ETSB should be requested to review the model.

The current spiking model assumes that at the time of tube failure the iodine release rate from the fuel rods to the primary coolant (expressed in unit of curies per unit time) increases by a factor of 500 relative to the release rate calculated assuming that the primary coolant iodine concentration is at the equilibrium concentration technical specification limit.

For the case with an iodine spike which already exists (due to a previous transient), the iodine concentration is assumed to be that allowed by Figure 3.4-1 of References 5, 6 or 7, or that which is proposed in Chapter 16 of the SAR.

7. Review of the effects of possible fuel damage. Additional coolant activity may become available for release if fuel failures result from the accident. The RSB reviews the effect of a steam generator tube failure (with the most reactive control rod stuck at its fully withdrawn position) on core thermal margins. If this event is predicted to cause fuel failures, RSB notifies AAB. (The reviewer assumes that the applicant's calculations of fuel damage are correct for the case of a control rod held at the fully withdrawn position unless informed otherwise by RSB.) If fuel damage does occur, calculations should be performed in order to assure that 10 CFR Part 100 guidelines are not exceeded (without a preaccident iodine spike).

8. Determination of the leakage into the unaffected steam generators. Normal operating primary-to-secondary leakage is assumed to exist in the unaffected steam generators. The leakage rate should be the maximum allowed by the technical specifications. Currently this value is 1 gpm. However, a lower value may be needed to limit the consequences of other events such as a control rod ejection accident or an anticipated transient without scram (ATWS).

9. Determination of the coolant flow through the failed tube. The leak rates through the two ends of the failed tube are calculated using a suitable flow model, taking credit for critical flow where appropriate.

10. Determination of the parameters for the thyroid dose calculation. The iodine transport model to be used is as described in Reference 2. A fraction of the iodine in the primary coolant escaping to the secondary system is assumed to become airborne immediately due to flashing and atomization. Credit may be given for "scrubbing" of iodine contained in the steam phase and in the atomized primary coolant droplets suspended in the steam phase for release points which are below the steam generator water level. That fraction of the primary coolant iodine which is not assumed to become airborne immediately enters the secondary system water and is assumed to become airborne at a rate determined by the steaming rate and iodine partition coefficient. An iodine partition coefficient of 100 between steam generator water and steam phases may be conservatively assumed unless the applicant presents reasonable evidence that the use of some other value is justified.

11. Calculation of the exclusion area and low population zone boundary doses. The reviewer computes the doses for the steam generator tube failure accident, both with and without preaccident iodine spiking. A breathing rate of 3.47×10^{-4} m^3/sec is used in the calculation of thyroid doses for the first 8 hours following the steam generator tube failure.

12. Review of the results of the dose calculations. The calculated doses are considered acceptably low if: (1) the doses calculated without assuming the existence of a pre-accident iodine spike, but assuming that iodine spiking occurs as a result of the accident, are less than a small fraction of 10 CFR Part 100 guidelines; and (2) the doses calculated assuming preaccident iodine spiking or fuel failures occur as a result of assuming that the most reactive control rod remains in its fully withdrawn position are less than the dose guidelines of 10 CFR Part 100. If the doses are not within these guidelines, the technical specification limits on equilibrium and/or spiked primary coolant activity should be reduced until the calculated doses are within the guidelines.

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 at the operating license stage:

"The steam generator tube failure accident has been evaluated with and without a concurrent loss of offsite power. The design of the plant has been found to assure an expeditious recovery from these accidents with acceptably limited activity releases.

"The assumptions used in our aralysis are listed in Table _____. The calculated doses are presented in Table ____. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within the 10 CFR Part 100 exposure guidelines even if the accident were to occur with a preaccident iodine spike or assuming additional fuel failures occur during the accident as a result of the most reactive control rod remaining fully withdrawn."

The following paragraph is added if fuel damage is found to be a possible consequence of the accident:

"The evaluation of the steam generator tube failure accident has been evaluated with ____% fuel damage in the core (as a result of the most reactive control rod remaining fully withdrawn). The resulting doses are within the guidelines of 10 CFR Part 100."

At the construction permit stage, the following paragraph is included in the staff's safety evaluation report:

"On the basis of our experience with the evaluation of steam generator tube failure accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations so that potential offsite doses are small. We will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications."

If the plant is a standard plant which has been reviewed before, the following paragraph may be used:

"The radiological consequences of a steam generator tube failure in the (vendor's name) standard NSSS/(A/E's name) BOP were analyzed generically by the staff, and the results reported in the staff SER dated _____. The offsite doses were found to be acceptable for sites with X/Q's equal to or less than _____ sec/m^3, if the coolant activity is limited to the values in the (vendor's name) Standard Technical Specifications (NUREG-_____, issue date). The technical specifications the applicant will use include these coolant activity limits. The staff has estimated the 0-2 hour X/Q at the exclusion area boundary of the site is _____ sec/m^3, as discussed in Section (2.3.4) in this SER. This X/Q is lower than the limiting X/Q for this standard NSSS/BOP plant. Therefore, we conclude that the offsite doses from a steam generator tube failure in the _____ plant would be less than current dose guidelines and are acceptable, although a specific dose calculation for this accident has not been performed.

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. A. K. Postma and P. S. Tam, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, USNRC, January 1978.

3. R. R. Bellamy, "A Regulatory Viewpoint of Iodine Spiking During Reactor Transients," Trans. Am. Nucl. Soc., 28 (1978).

4. W. F. Pasedag, "Iodine Spiking in BWR and PWR Coolant Systems," CONF-770708, 3-217 (1977).

5. Standard Technical Specifications for Combustion Engineering PWRs, NUREG-0212.

6. Standard Technical Specifications for Westinghouse PWRs, NUREG-0452.

7. Standard Technical Specifications for Babcock and Wilcox PWRs, NUREG-0103.

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SECTION 15.6.4 RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Hydrology-Meteorology Branch (HMB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The purposes of the review are to calculate the whole body and thyroid doses resulting from a postulated failure of a main steam line outside containment, and to assure that releases due to the failure are adequately limited by the technical specifications on primary coolant activity. The RSB will notify the AAB if this accident is predicted to cause fuel failures. The RSB may also be requested to confirm the assumption of instantaneous reactor coolant release. The HMB provides the reviewer with the appropriate X/Q values for this event.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against the radiological consequences of a main steam line failure outside containment, and the primary coolant activity appropriately limited by the technical specifications if calculations show that the resulting doses at the exclusion area boundary are small fractions (equal to or less than 10%) of the 10 CFR Part 100 exposure guidelines.

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.

At the construction permit (CP) stage, the review is limited to a brief survey of the pertinent portions of the plant design and the applicant's discussion of the accident to determine that there are no unusual features that would prevent limitation of doses to acceptable levels by appropriate limits on coolant activity concentrations. Standard

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Technical Specifications regarding coolant activity concentration limits have been issued for BWR plants and the radiological consequences of a steam line failure have been evaluated for a standard GESSAR plant using this activity limit to determine the limiting X/Q. Consequently, the radiological consequences of a steam line failure accident need not be explicitly calculated for a standard GESSAR plant located at a site where the X/Q value is equal to or less than the limiting X/Q value.

The detailed review of the radiological consequences of a main steam line failure outside containment is done at the operating license (OL) stage when system parameters and accident analysis are fully developed.

The AAB review of main steam line failure accidents at the OL stage consists of the following steps:

1. Review of the applicant's descriptions of the steam line failure accidents.

2. Performance of an independent analysis of the radiological consequences of the failure of the main steam line, using the assumptions other than atmospheric diffusion of Regulatory Guide 1.5 (Ref. 2). At present, the following conservative assumptions are used to simplify the analysis:

   a. The mass of reactor coolant assumed to be released to the environment is 140,000 pounds for the "251" sized BWR and 100,000 pounds for the "238" sized BWR. Other BWRs should be evaluated on a case-by-case basis. The release is assumed to occur instantaneously. (These assumptions are made unless notified otherwise by RSB.)

   b. The initial fission product concentrations in the primary coolant are those given as the maximum values in the technical specifications. The RSB reviews, on a generic basis, the effect on core thermal margins of a steam line failure in a BWR. If this event is predicted to cause fuel failures, RSB notifies AAB so that the predicted magnitude and extent of fuel failures can be properly considered in the evaluation of the radiological consequences. No decontamination factors, or other reductions in the fission product concentrations are assumed.

3. The HMB provides the reviewer with the appropriate X/Q values for this event.

4. Review of the results of the dose calculations. The resulting doses from these accidents should be small fractions (equal to or less than 10%) of the 10 CFR Part 100 limits. If this is not the case, the primary coolant concentration limits in the technical specifications should be reduced.

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 at the operating license stage:

"The applicant's analysis of a failure of a main steam line has been reviewed, and an independent analysis of this accident has been performed by the staff. The results of the staff's calculations are presented in Table ____. The technical specifications for the primary coolant activity concentrations limit the consequences of this accident to a small fraction of 10 CFR Part 100 exposure guidelines."

At the construction permit stage, the following paragraph is included in the staff's safety evaluation report:

"On the basis of our experience with the evaluation of steam line failure accidents for boiling water plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary coolant radioactivity concentrations so that potential offsite doses are small. We will include appropriate limits on the primary coolant activity concentrations in the technical specifications."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."

SECTION 15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Auxiliary Systems Branch (ASB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Instrumentation and Control Systems Branch (ICSB)
Mechanical Engineering Branch (MEB)
Analysis Branch (AB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

Loss-of-coolant accidents (LOCA) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and to include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished.

Each pressurized water reactor (PWR) and boiling water reactor (BWR) must be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS given in 10 CFR 50.46 and Appendix K to Part 50 (Ref. 1) and the applicable general design requirements discussed in SRP Section 6.3 (Ref. 2). The analysis of ECCS performance has an impact on the design of the piping and support structures for the reactor coolant system, the design of the steam generators, the containment design, and the possible need for pump overspeed protection.

The review of the applicant's analysis of the spectrum of postulated loss-of-coolant accidents is closely associated with the review of the ECCS, as described in SRP Section 6.3. As a portion of the review effort described in this SRP section and in SRP Section 6.3, RSB evaluates whether the entire break spectrum (break size and location) has been addressed; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA

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for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on ECCS performance have been provided. For postulated break sizes and locations, the RSB review includes the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

The calculational framework used for the evaluation of the ECCS system in terms of core behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. The evaluation model used by the applicant must comply with the acceptance criteria for ECCS given in 10 CFR 50.46 and Appendix K to Part 50. The evaluation model must have been previously documented and reviewed and approved by the staff. Should the LOCA blowdown calculations be modified for the purpose of studying structural behavior (for example, core support structure design, control rod guide structure design, steam generator design, reactor coolant system piping and support structure design), all differences should be identified and described by the applicant. On request, RSB reviews these modifications, including analytical techniques, computer programs, values of input parameters, break size, type, and location, and all other pertinent information, and makes recommendations regarding their acceptability to other branches as required. RSB requests generic computer code reviews from AB as required.

RSB is also responsible for the review of the failure mode analysis of the ECCS in conjunction with the effort described in SRP Section 6.3. ASB, PSB and ICSB provide assistance in this review, on request.

AAB provides an evaluation of fission product releases and radiological consequences. This effort is described in the appendices to this SRP section.

ASB, as described in the plans for SAR Chapters 9 and 10, provides an evaluation of auxiliary systems (e.g., service water system, component cooling system, ultimate heat sink, condensate storage facility) to confirm that these systems can supply all the functions required to support the ECCS in performing its function during and following a loss-of-coolant event. ASB also, on request from RSB, reviews the failure mode analysis of the ECCS.

CSB, as described in SRP Section 6.2.1, evaluates the functional capability of the containment for the spectrum of loss-of-coolant events. The assumptions used for the containment

response analysis must be selected in a manner conservative for the purpose. CSB, on request from the RSB, also provides an evaluation of containment pressure calculations utilized in the reflood portion of the ECCS performance analyses.

CPB, upon request, reviews the power transient calculations including moderator temperature, void and fuel temperature feedback effects, and decay heat. AB, upon request, reviews the analytical techniques used for blowdown, reflood, and clad temperature calculations; and performs independent blowdown, reflood, and clad temperature calculations, as described in SRP Section 4.3 and the appendix to SRP Section 4.4.

ICSB, as described in SRP Sections 7.2, 7.3, 8.3.1 and 8.3.2, reviews the protection system and ECCS-associated controls and instrumentation with regard to automatic actuation, remote sensing and indication, remote control, and redundancy. PSB evaluates the emergency onsite power functional capabilities.

ICSB and PSB also, upon request, review the failure mode analysis related to the instrumentation and electrical power supply submitted by the applicant to show that the most damaging single active failure of the ECCS was selected for the LOCA analysis.

MEB, as described in SRP Sections 3.6 and 3.9, is responsible for the review of the effects of blowdown loads on core support structures and on control rod guide structures. MEB verifies that the core remains in place in case of a LOCA and that the control rods can be inserted. MEB is also responsible for evaluating the effects of blowdown loads including jet forces on the piping of the reactor coolant system and on the support structures of the components of the reactor coolant system. MEB verifies that acceptable criteria have been employed in the design of the reactor coolant system and its supports to prevent failures in the reactor coolant pressure boundary or in engineered safety feature equipment in the event of a LOCA.

II. ACCEPTANCE CRITERIA

The objectives of the review of the applicant's analysis of loss-of-coolant accidents are to verify that:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an approved evaluation model that satisfies the requirements of 10 CFR 50.46 and Appendix K to Part 50. For the full spectrum of reactor coolant pipe breaks, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied as given below:

   a. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

   b. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

d. Calculated changes in core geometry are such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100.

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 both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

For the review of the ECCS performance analysis, as presented in the applicant's safety analysis report (SAR), the reviewer verifies the following:

1. The calculations were performed using an approved evaluation model. The application should clearly state this and properly reference the evaluation model. If the analysis is done with a new evaluation model, a generic review of the new model is required.

2. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure. This analysis is reviewed in part under SRP Section 6.3. If the design has been changed from that presented in previous applications, changes in the reactor coolant system, reactor core, and ECCS are reviewed with respect to the most limiting single failure.

3. A variety of break locations and the complete spectrum of break sizes were analyzed. If part of the evaluation is done by referencing earlier work, design differences (ECCS, reactor coolant system, reactor core, etc.) between the facilities in question are reviewed. If there are significant differences, sensitivity studies on the important parameters should have been made by the applicant. If such sensitivity studies are not presented in the SAR, the reviewer requests that they be made.

4. The parameters and assumptions used for the calculations conform to those of the approved evaluation model and were conservatively chosen, including the following points:

   a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

   b. The maximum linear heat generation rate used should be based on 102% of the proposed licensed core thermal power and the technical specification limit on peaking factors, or on the technical specification limits on maximum linear heat generation rate.

   c. All permitted axial power shapes, as given in Section 4.3 of the SAR, should be addressed by the analyses. Normally, the evaluation model will identify the least favorable axial shape as a function of break size. If the evaluation model did not discuss axial shapes, or the discussion is not applicable to a given case, sensitivity studies are requested.

   d. The initial stored energy was conservatively calculated by the applicant. The value used is checked against the applicant's steady-state temperatures, as given in SAR Section 4.4, similiar calculations performed by the staff, or calculations done for similar plants by previous applicants.

   e. Appropriate analyses are presented to support any credit taken for control rod insertion.

5. Reactor protection system actions and safety injection actuation and delivery are consistent with the set points and the associated uncertainties and delay times listed in the SAR (OL review). The ECCS flow rates should be checked against the applicant's data on head-flow characteristics of the ECCS pumps given in Section 6.3 of the SAR and against typical safety injection tank discharge curves used for the analysis. The Regional Offices under the Office of Inspection and Enforcement may be requested to provide data of this type from the startup tests for new designs and from periodic tests on duplicate designs.

6. The results of the applicant's calculations are consistent with those of staff calculations for typical plants and also with the results of calculations performed for similar systems by previous applicants. The following variables should be reviewed on a generic basis and spot-checked thereafter: power transients for various breaks; pressure transients at various system locations; flow transients near the break, in the core, and in the downcomer; reactor coolant temperature and quality at core

inlet, core outlet, and in-core; cladding temperature transients (core average, hot assembly, hot pin); heat transfer coefficients during blowdown, refill, and reflood; heat flux transients from piping and vessel walls; primary-secondary heat transfer (PWRs only); timing of clad rupture (if the peak clad temperature could be appreciably higher when perforation occurs at a different but equally probable time, calculations with modified assumptions are requested); peak clad temperature as a function of break size (if it is uncertain whether the peak value has been found, additional calculations are requested); predicted "end-of-bypass" time compared to calculated downcomer flow and to staff calculations for typical plants; pump speed transients; containment pressure transients (if staff calculations are not available, these are requested from CSB); and carryover fraction (if it is not an input to the calculations).

7. The calculated peak clad temperature, maximum local oxide thickness, and core average zirconium-water reaction meet the acceptance criteria for ECCS given in 10 CFR 50.46 and Appendix K to Part 50.

8. The applicant's analysis addresses the full LOCA sequence of events to the point where the plant is in the long-term cooling mode and removal of decay heat has been well established for both large and small breaks. The reviewer checks the assumed sources of coolant water, redundancy of delivery routes, alignment of valves, control of boron concentration (PWR) and all required operator actions.

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.

The review of fission product releases and radiological consequences of design basis (most severe) LOCA is performed by AAB as described in the appendix to this SRP section.

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 applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR § 50.46 and Appendix K to Part 50). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff as described in _____. The results of the analyses show that the ECCS satisfy the following criteria:

1. The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

2. The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry are such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

"The radiological consequences of the postulated spectrum of loss-of-coolant accidents (LOCA) were evaluated from the viewpoint of site acceptability. For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS."

The evaluation findings of the AAB resulting from the reviews detailed in Appendices A, B, C, and D, as applicable, should be inserted in the safety evaluation report draft at this point. See Appendices A - D for typical findings and conclusions.

"The staff concludes that the calculated performance of the emergency core cooling system following a postulated loss-of-coolant accident and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, accordingly, the ECCS is considered acceptable."

V. REFERENCES

1. 10 CFR § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and Appendix K to Part 50, "ECCS Evaluation Models."

2. Standard Review Plan Section 6.3, "Emergency Core Cooling System."

APPENDIX A

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: CONTAINMENT LEAKAGE CONTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

1. The review is concerned with the selection of the values of plant parameters used in calculating the radiological consequences of containment leakage following a loss-of-coolant accident. It is also concerned with selecting a dose computation model that incorporates conservative transport mechanisms and rates from various parts of the containment to the atmosphere, suitable breathing rates, dose conversion factors, and other physical and biological data that may affect the computed dose.

2. The calculated doses are compared with the appropriate exposure guidelines to confirm the acceptability of the nearest exclusion area boundary and low population zone (LPZ) outer boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

The ETSB reviews ESF filter system design and filter efficiencies in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

1. The fractions of the fission product inventory assumed to be available for release from the containment are acceptable if they agree with the values listed in Section C of Regulatory Guide 1.3 or Regulatory Guide 1.4. No specific list of isotopes or decay constants has been selected as standard.

Where the applicant claims a single containment system, this is accepted. To receive credit for a dual containment system, a determination must be made that the system meets the necessary requirements. These requirements are detailed in Standard Review Plan (SRP) 6.2.3 and SRP 6.5.3. Containments falling outside of these categories are evaluated on a case-by-case basis. For single containment systems, or leakage from the primary containment, the leakage rate stated in the technical specifications is accepted subject to verification by the CSB, provided a leakage rate of at least 0.1% per day is stated. For a boiling water reactor (BWR) the leakage rate is currently assumed constant over the course of the accident, while for a pressurized water reactor (PWR), the leakage rate is reduced after 24 hours to one-half its original value (see Refs. 2 and 3). Where a single containment is specified, no credit for

exhaust filters is allowed, although internal recirculation filters can be credited, if present.

2. The methods used to calculate radiological consequences of a postulated LOCA are acceptable if they reflect the use of conservative design basis assumptions as outlined in Regulatory Guide 1.3 or 1.4 (Refs. 2 and 3). The requirements of 10 CFR Part 100 are that the total dose from a postulated loss-of-coolant accident (LOCA) to an individual (located at positions specified in 10 CFR Para. 100.11(a)) must be no greater than 300 rem to the thyroid and 25 rem to the whole body. At the construction permit (CP) stage, exposures of no more than 150 rem to the thyroid and 20 rem to the whole body are considered acceptable to allow for uncertainties in meteorology and other site-related data and to allow for system design changes that might influence the final design of engineered safety features or the dose reduction factors allowed for these features. This lower guideline is required at the CP stage to provide reasonable assurance that the 10 CFR Part 100 guideline values can be met at the operating license (OL) review stage.

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 it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The design (stretch) power level of the core is taken from the applicant's safety analysis report (SAR). The core is assumed to have operated at this power level for a sufficiently extended period (typically about 3 years) that a maximum equilibrium fission product inventory is present. At time of the accident, 25% of all the equilibrium iodine fission products and 100% of the noble gas fission products are assumed available for release from the containment within a very short time (effectively instantaneously) after the accident. The iodine is assumed to be composed of 91% elemental iodine, 4% organic iodides, and 5% particulate iodine.

2. From the applicant's SAR (parts of Section 1, Sections 6.2.1, and 6.2.3), the reviewer ascertains the type of containment system used. A check is made of the LOCA assumptions listed in Chapter 15 of the SAR to verify that the primary containment leakage rate has been assumed to remain constant over the course of the accident (for a BWR) or to be halved after 24 hours (for a PWR) and that the initial leak rate is at least 0.1% per day (a lower limit is set because of integrated containment leakage test sensitivity limitations). The leakage rate used should correspond with that given in the technical specifications in SAR Chapter 16.

3. Where credit for a dual containment system is claimed, the reviewer verifies (see SRP 6.2.3 and 6.5.3) that the system meets requirements such as existence of separate primary and secondary containments, adequate separation of the two, and ability to test the negative pressure capability of the secondary containment volume. Where credit

for a secondary containment with recirculation is claimed, adequate mixing in the secondary containment volume should be demonstrated in addition to meeting the above requirements for a dual containment system. For dual containment systems, the bypass leakage rate is noted. This leakage, usually expressed as a fraction or percentage of the primary containment leak rate, is assumed to go from the primary containment directly to the environment, bypassing the secondary containment. This bypass leakage rate, as well as any positive pressure conditions should be verified by the CSB. See SRP 6.2.3 for a detailed treatment of bypass leakage.

4. Credit, if any, to be given for any engineered safety features such as filters, sprays, or ice condenser that may be present, is determined in the review of Section 6.5 of the SAR. These features operate during the LOCA to mitigate the consequences by reducing the amount of iodine fission products released to the environment. Noble gas releases to the environment are unaffected by the presence of filters or sprays. Typically, single containments employ spray systems with a chemical additive (e.g., sodium hydroxide, sodium tetraborate) designed to scavenge iodine from the containment atmosphere. The iodine removal rates of an ice condenser or a chemical additive spray system are determined after consultation with specialists in this area. For filters, verification of acceptability of design and filter efficiencies is provided by the ETSB in SRP 6.5.1. In dual containment systems, a determination must be made by the AAB of the operational modes of the ESF with respect to the accident sequence in order for proper credit to be given.

5. Sections 2.1.2 and 2.1.3 of the applicant's SAR are examined to determine the minimum distances to the exclusion area boundary and to the LPZ outer boundary. Following the procedures given in SRP 2.1.2 and SRP 2.1.3, the reviewer confirms the validity of the applicant's values. From the SAR, the reviewer also obtains relevant information (e.g., locations and time durations) concerning activities unrelated to plant operation that may exist inside the exclusion area boundaries (see SRP 2.1.2). In some cases specific dose computations may have to be performed to assist in determining the adequacy of evacuation plans.

6. The SAB is requested to furnish suitable X/Q values to be used in analyzing the consequences of the accident. X/Q values are obtained not only at the nearest exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion boundary where significant activities may occur involving members of the public.

7. Based upon the review procedures already performed, a dose computation model is selected which conservatively represents the transfer of radioactivity from the containment to the environment. The reviewer may find it convenient to sketch a schematic arrangement to illustrate the compartments where radioactivity is located, with arrows drawn from one compartment to another indicating transport paths. The leak rates, spray removal rates, ice condenser efficiencies, filter efficiencies, and flow rates are all used to indicate the rates at which the activity moves from one compartment to another. Digital computer codes (Ref. 4) have been written to perform

the actual dose calculation. The analyst should select the code with capabilities that most closely fit the schematic model obtained above. The codes contain a basic library of physical and biological data which enter into the dose calculation, such as isotopic fission yields, half-lives, energies, and dose conversion factors.

8. The calculated doses, including the 2-hour thyroid inhalation and whole body doses at the nearest exclusion area boundary, the thyroid inhalation and whole body doses for the course of the accident at the outer boundary of the LPZ, and those doses calculated at other points within the exclusion area boundary where there may be activities unrelated to to plant operation at certain times, are compared with the dose guidelines as discussed in Section II.2 of this plan. Where the results of the dose calculations exceed the guidelines, the alternatives which would reduce the doses to an acceptable level are explored (e.g., increased distance, secondary containment, better filter or spray systems). The feasibility of the alternatives is also examined. The AAB Branch Chief is consulted as to appropriate action in this case.

IV. EVALUATION FINDINGS

If the AAB reviewer finds that the radiological consequences of the containment leakage contribution to a loss-of-coolant accident are acceptable, conclusions of the following type may be included with the RSB findings for this area in the staff's safety evaluation report:

"The radiological consequences of a loss-of-coolant accident as a result of leakage from the containment were evaluated. The analysis of the containment leakage doses following a postulated design basis loss-of-coolant accident included the influence of fission product removal and holdup systems and the containment leakage routes on the estimated radiological consequences.

"The review has included the applicant's proposed design criteria and design bases for the effect of containment leakage and his analysis of the manner in which the containment leakage consequences conform to the design criteria.

"The basis for acceptance in the staff review has been confirmation that the applicant's analysis conforms with the applicable regulations, regulatory guides, technical positions and industry standards as listed in Table 15.13.2-1. The staff concludes that the proposed design, including leakage rates and fission product removal and control systems conform to the Commission's regulations and to applicable regulatory guides and staff technical positions, and that the conservatively computed doses from containment leakage following a loss-of-coolant accident are within the exposure guidelines of 10 CFR Part 100."

Appropriate tables of assumptions used and the estimated consequences are to be included in the SER. The following should be added at the CP stage: "Because the proposed design meets the recommendations of Regulatory Guide 1.3(1.4), there is reasonable assurance that the exposure guidelines of 10 CFR Part 100 can be met at the OL stage."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2.

3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2.

4. Computer codes are currently under development. Documentation will be published in a NUREG report.

APPENDIX B

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: LEAKAGE FROM ENGINEERED SAFETY FEATURES COMPONENTS OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

A potential source of fission product leakage following a loss-of-coolant accident (LOCA) is the leakage from engineered safety features (ESF) equipment which is located outside the primary containment. Such leakage could occur during the recirculation phase for long-term core cooling and primary containment (spray) cooling. The total leakage from these sources is added to that resulting from the containment leakage following a LOCA. To calculate the maximum potential leakage from the recirculation loop, such sources as the following are considered: containment spray system, low pressure safety injection system, and high pressure safety injection system.

The ETSB reviews ESF ventilation system filters for conformance with Regulatory Guide 1.52 in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

The source of leakage is related to the requirement to detect and isolate failures of passive components in the long-term (recirculation) mode for ESF systems. Therefore, leakage anywhere in the systems carrying recirculation water outside of containment is postulated. ESF-grade filtration systems to process potential leakage are required as the dose could exceed 10 CFR Part 100 guidelines without filters even at relatively low leakage rates resulting from passive failures. When ESF-grade filters are supplied, no doses resulting from passive failures need be considered.

The acceptance criterion for the dose resulting from leakage outside primary containment from the recirculation systems is that when it is added to the dose attributable to containment leakage, including any main steam isolation valve sealing system leakage (Appendices A and D of Standard Review Plan 15.6.5), the total dose is to be within the guideline values of 10 CFR Part 100. To provide assurance that this criterion is met at the operating license stage, the doses indicated in Regulatory Guides 1.3 and 1.4 are used as acceptance criteria at the construction permit stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review 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 if it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The applicant's recirculation leakage calcualtion is checked against previously licensed plants for accuracy and completeness. It is assumed that 50% of the core iodine inventory, based upon the maximum reactor power level, is mixed in the sump water being circulated through the external piping systems. Credit may be allowed for radioactive decay of the iodine during the time period from the occurrence of the LOCA up to the beginning of recirculation when the sump water is circulated outside the containment.

The dose computed for presentation in the staff safety evaluation report (SER) should be based upon twice the maximum operational leakage and should be assumed constant for the course of the accident. The maximum operational leakage is defined as the sum of the leakage for all the recirculation systems (1) which is detectable during test and (2) above which the technical specifications whould require declaring a system out of service. The leakage is assumed to occur throughout the accident, starting at the earliest time that recirculation mode is initiated.

The applicant's data on sump water temperature versus time after the LOCA should be consulted and used. During the time that the circulating water temperature exceeds 212°F, the fraction of water flashing to steam should be computed and taken as the fraction of iodine in the water which becomes volatile. In those cases where the circulating water temperature is less than 212°F, 10% of the iodine in the water which leaks is assumed to become volatile unless a smaller amount is justified based on actual sump pH history and ventillation rates.

All the iodine becoming volatile is assumed to be released immediately to the environment, and atmospheric dispersion is based upon the ground level X/Q values determined by the SAB. Any ventilation system filters are evaluated by the ETSB for compliance with Regulatory Guide 1.52 (Ref. 4) and appropriate credit for iodine removal by the filters given. The nearest exclusion area boundary and LPZ outer boundary doses are calculated by standard methods as described in Appendix A to SRP 15.6.5.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings for Section 15.6.5 in the SER:

> "The staff concludes that doses resulting from the postulated leakage of post-LOCA recirculation water from pump seals, valve packings, etc., are low and, when added to the direct leakage LOCA doses, result in total doses that are within the guideline values of 10 CFR Part 100. Engineered safety feature-grade filtration systems are provided to process potential leakage from postulated failures of passive components in systems carrying post-LOCA recirculation water outside of containment."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2.

3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2.

4. 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."

5. Appendices A and D, Standard Review Plan 15.6.5.

APPENDIX C

STANDARD REVIEW PLAN SECTION 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: POST-LOCA PURGE CONTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)
Hydrology-Meteorology Branch (HMB)

I. AREAS OF REVIEW

AAB reviews the radiological consequences of purging the containment after a postulated loss-of-coolant (LOCA) to establish that the LOCA-plus-purge doses are acceptable and, in some cases, to determine whether additional filtration systems are needed. CSB reviews the applicant's proposed purge system for design acceptability under criteria listed in Reference 2. ETSB reviews the applicant's filter efficiencies for the purge system filters (if any). HMB determines the ground level release X/Q for the time period specified by AAB.

II. ACCEPTANCE CRITERIA

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.

The plant is considered adequately designed against the radiological consequences of post-LOCA containment purge if the following criteria are met.

For plants for which a notice of hearing on the application for a construction permit was published after November 5, 1970:

> Purging shall not be the primary means of controlling combustible gases following a LOCA. However, all water-cooled power reactors should also have the installed backup capability for a controlled purge of the containment atmosphere. Unless unusual circumstances dictate or unless requested by the CSB, an incremental post-LOCA purge dose calculation for the backup purge system will not be performed.

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:

The incremental long-term doses from purging in the event of a postulated LOCA should be less than 2.5 rem whole body and 30 rem thyroid at all points beyond the exclusion area boundary. 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.

For plants for which a notice of hearing on the application for a construction permit was published before December 22, 1968:

The sum of the long-term doses from a postulated LOCA and the purging dose should be below the guidelines of 10 CFR Part 100 at the low population zone outer boundary.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the area covered by this SRP Section as may be appropriate for each 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.

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.

AAB and CSB review the proposed purge system design and determine whether a purge dose calculation should be performed. The analysis should be performed for cases (post-LOCA) which result in additional doses from purging or for cases when the purge system is relied upon (as the primary means) to control the containment atmosphere quality following an accident. If the analysis is required, the reviewer will perform a purging dose calculation following the procedures outlined below.

1. Source Terms

   a. Iodine

      The initial airborne iodine-131 component is assumed to be 25% of the core inventory and the iodine airborne activity at any subsequent time is subject to a removal factor due to the plant engineered safety features and to radioactive decay as stated in Regulatory Guides 1.3 and 1.4.

   b. Noble Gases

      The initial Xe-133 and Kr-85 activities are assumed to be 100% of the core inventory, as stated in Regulatory Guides 1.3 and 1.4. These nuclides are subject to removal through radioactive decay only.

2. Purging

   The model assumes a constant purge rate after initiation of purge. No credit is included for depletion of activity due to containment leakage.

When a dose calculation is required for a plant, the data required for this calculation are obtained from the following sources. AAB obtains the LOCA analysis assumptions as to reactor power level, primary containment volume, and iodine reduction factors from Appendix A to Standard Review Plan Section 15.6.5. The AAB reviewer is responsible for transmitting the requirements for filters to the ETSB when such requirements are indicated by the results of the dose calculation. CSB determines the purge rate in scfm and the holdup time (in days) prior to purging. ETSB determines filter efficiencies in cases where filters are required to meet the dose criteria. HMB determines the ground level release X/Q derived from onsite data for the time period specified by AAB.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings for Section 15.6.5 in the staff's safety evaluation report:

> "The analysis of the radiological consequences of containment purging following a LOCA yields thyroid and whole body doses which are within the acceptance criteria of SRP Section 15.6.5, Appendix C and the combined purge plus LOCA dose is less than the guideline values of 10 CFR Part 100 Acceptance Criteria.

If the reviewer finds the consequences unacceptable, then the following may be stated for current reviews:

> "The analysis of the radiological consequences of containment purging following a LOCA indicates that the total long-term doses from the LOCA and the purge exceed the guideline values of 10 CFR Part 100 at the LPZ outer boundary. Accordingly, remedial measures (inert gas injection or filters) are required to achieve acceptable dose levels."

Conclusions which match the acceptance criteria for older plants should be drafted for such plants.

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Proposed Rule 10 CFR 50-44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors."

3. 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."

4. Appendix A, Standard Review Plan Section 15.6.5.

APPENDIX D

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: LEAKAGE FROM MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Site Analysis Branch (SAB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

A potential source of fission product leakage following a loss-of-coolant accident (LOCA) is the leakage past the main steam isolation valves in a BWR. This leakage is required to be controlled by a main steam isolation valve leakage control system (MSIVLCS). This system may act as a positive sealing system or a vacuum-type system which collects leakage between the closed isolation valves and releases it to the atmosphere through a filter system. The method of operation, time of operation, and release paths associated with the operation of the MSIVLCS are reviewed to calculate the fission product releases and their contributions to the doses at the nearest exclusion area boundary and LPZ outer boundary. Any leakage from the isolation valves (e.g., valve stem leakage) or any release from the MSIVLCS is added to the containment leakage and ESF leakage (Appendices A and B of Standard Review Plan 15.6.5) following a LOCA.

II. ACCEPTANCE CRITERIA

The calculated doses associated with operation of the MSIVLCS following a postulated LOCA should be limited so that when they are added to the dose contribution from containment leakage and leakage from ESF components outside containment, the total does not exceed the guideline values of 10 CFR Part 100 (Ref. 1) at the operating license stage or Regulatory Guide 1.3 (Ref. 2) at the construction permit stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review 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 if it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The applicant's description of the MSIVLCS is reviewed to familiarize the reviewer with the system performance and to obtain the information needed to perform the dose calculation. For a positive sealing system, verification of the system operability assuming a single active failure, actuation time, and identification of any potential release paths is obtained

from the APCSB. If the reviewer finds that no release paths exist and that the system can be actuated within an appropriate time after the accident, no further review is required.

For a vacuum-type system, which processes rather than seals the leakage, the following information, assuming the most adverse single failure of an active component, must be verified by the APCSB (and documented by buckslip to the AAB):

1. Release paths and fractions of the leakage through these paths, as a function of time, e.g., steam leakage, releases through a depressurization line, releases through drain lines, etc.

2. System actuation time.

3. Flow rates as a function of time.

4. Release points.

Interaction with systems used to mitigate the consequences of containment leakage should be noted. It may be necessary to establish with the CSB that the operation of the MSIVLCS does not adversely affect pressure transients in secondary containment regions.

The system is then modeled using a computer code (Ref. 3). The source assumed is the same as that used to estimate the containment leakage dose, but it is assumed to be instantaneously distributed in the drywell free volume at the time of the accident. Credit for decay in the drywell is given; no release is assumed up to the time of system actuation; but no credit is given for leakage from the drywell to the containment (Mark III) or the suppression pool region (Mark I and II). The main steam isolation valves are assumed to be leaking at their technical specification limit. Leakage through valve stems or drain lines to an untreated region is assumed to be released to atmosphere; releases through the MSIVLCS which are directed to treated regions are assumed to be directly to the filter intake unless the MSIVLCS flow is mechanically directed to a distributed header. If the latter is the case, then credit for mixing is given on the same basis as in other leakage to this system (see Standard Review Plan 6.5.3).

The resulting doses are calculated using the model described in Regulatory Guide 1.3 (Ref. 2). The X/Q values to be used are the accident X/Q's provided by the SAB. For systems which are designed for initial releases at later times into the accident, application of worst meteorology at the time of release may have to be considered; this will be handled on a case-by-case basis.

The doses are added to those estimated from containment leakage and the leakage from ESF components outside containment and the total is compared to the guidelines of 10 CFR Part 100 (Ref. 1) if the application is for an operating license and to the guidance of Regulatory Guide 1.3 (Ref. 2) if the application is for a construction permit.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings for Section 15.6.5 in the staff's safety evaluation report:

> "The radiation doses resulting from main steam isolation valve leakage and operation of the main steam isolation valve leakage control system following a postulated LOCA were estimated assuming a single failure that is most adverse from the standpoint of radiological consequences. The analysis included the influence of fission product removal systems, delay times, and various release paths. The review has established that the applicant's design is sufficient to limit the radiological consequences due to the main steam isolation valve leakage or due to operation of the MSIVLCS such that when combined with the releases from other paths, the total potential consequences at the nearest exclusion area boundary and at the low population zone outer boundary are well within the guidelines of 10 CFR Part 100."

Appropriate tables of assumptions used and the estimated consequences are to be included in the SAR. The following should be added at the CP stage: "Because the proposed design meets the guideline values of Regulatory Guide 1.3, there is reasonable assurance that the exposure guideline of 10 CFR Part 100 can be met at the operating license stage."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2.

3. Computer codes are currently under development. Documentation will be published in a NUREG report.

4. Appendices A and B, Standard Review Plan 15.6.5.

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SECTION 15.7.1 WASTE GAS SYSTEM FAILURE

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

1. The radiological consequences of an unexpected and uncontrolled release to the atmosphere of radioactive fission gases that are stored or transferred in the waste gas system are reviewed to determine compliance with dose criteria.

2. The applicant's safety analysis report (SAR) is reviewed to see that there are technical specifications to limit to an appropriate level the activity which could be released, assuming the failure of any active component in the waste gas system. (The ETSB reviews the waste gas system design on SRP 11.3.)

II. ACCEPTANCE CRITERIA

Failure of waste gas systems that comply with the current staff position on seismic and quality group design requirements should result in doses well within the guideline values of 10 CFR Part 100 using conservative calculational assumptions. Systems whose failure would result in exposures approaching or exceeding the guideline values of 10 CFR Part 100 should have a technical specification limiting the amount of activity in the system. The technical specification should limit the activity in waste gas system components such that any single failure of a component will not result in a whole body dose at the nearest exclusion area boundary in excess of 0.5 rem.

For older plants, where it is proposed to modify the waste gas systems, systems which do not meet the current seismic requirements should have a technical specification such that the two-hour whole body dose at the nearest exclusion area boundary, in the event of a system failure, is less than 5 rem using conservative calculational assumptions.

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

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.

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.

If the current staff positions on quality and seismic design provisions are complied with, the probability of failure of passive components is sufficiently low that the consequences may be compared with the exposure guidelines of 10 CFR Part 100. Conservative calculational assumptions, similar to those used in analyzing other design basis accidents, should be used. (See, for example, Standard Review Plan 15.6.5.)

The reviewer obtains assistance from the ETSB as necessary to account for special characteristics of the waste gas system. The appropriate X/Q values are obtained from the SAB. The radiological consequences are then computed using a digital computer code (Ref. 2).

At the operating license stage, the reviewer verifies that a technical specification has been provided to limit the waste gas inventories in components such that single active failures of components, such as the lifting and sticking of a relief valve, will result in small exposures when the consequences are conservatively calculated. Unless otherwise demonstrated, all the inventory of a pressurized tank (and normally interconnected tanks if these would not be automatically isolated) is assumed to be released on lifting and sticking of a relief valve. If charcoal is present, a slow evolution of gas may be assumed for the fraction of gas shown not to be released during the tank depressurization.

IV. EVALUATION FINDINGS

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

> "Potential failures of the waste gas system have been reviewed and the computed doses have been found to be well within the guideline values of 10 CFR Part 100. Technical Specification limits have (will be) set to limit potential doses from single failures of active components to small fractions of the 10 CFR Part 100 guideliens."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."
2. Computer codes are currently under development. Documentation will be published as a NUREG report.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE (RELEASE TO ATMOSPHERE)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The radiological consequences of the release to the atmosphere of radioactive fission gases resulting from an unexpected and uncontrolled release of radioactive liquids that are stored or transferred in a waste system are reviewed by the AAB to determine that they are small fractions of the 10 CFR Part 100 guideline values.

The ETSB reviews the radioactive liquid waste system in SRP 11.2 and the liquid radioactive source term in SRP 11.1.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a radioactive liquid waste system leak or failure if the conservatively calculated exposures resulting from the release of radioactive gases from the system are small fractions of the 10 CFR Part 100 guideline values. Iodine decontamination factors similar to those specified in Standard Review Plan (SRP) 15.6.5, Appendix B, are acceptable.

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 to be 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.

If the major parameters of the applicant's calculation (e.g., iodine decontamination factors, direct release of gases to environment, and X/Q values) are in agreement with the staff's values and if the calculated doses are less than 0.5 rem to the whole body and 1.5 rem to the thyroid, no further review is required.

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If the reviewer finds that an independent calculation is necessary, the following steps are taken:

1. The appropriate atmospheric dispersion parameters are obtained from the SAB for use in calculating the thyroid and whole body doses.

2. Information describing an accident involving a radioactive liquid leak or tank failure, including the source term, is obtained from the ETSB. This should include a case with dissolved noble gases and one with high iodine inventory.

3. The thyroid and whole body doses resulting from gaseous transport from liquid spills are calculated using the above information and that presented by the applicant in the safety analysis report.

4. The calculations described above should take into account any related technical specifications and should be based on assuming an appropriate value for the iodine decontamination factor. (See SRP 15.6.5, Appendix B.)

5. The resulting doses are evaluated to determine that they are small fractions of the 10 CFR Part 100 guideline values.

IV. EVALUATION FINDINGS

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

"The radiological consequences of the release to the atmosphere of radioactive fission gases resulting from an unexpected and uncontrolled release of radioactive liquids that are stored or transferred in a waste system have been evaluated and the estimated doses found to be small fractions of the 10 CFR Part 100 guidelines values."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Standard Review Plan 15.6.5, Appendix B.

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SECTION 15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Hydrology-Meteorology Branch (HMB)

I. AREAS OF REVIEW

1. The ETSB reviews the consequences of single failures involving tanks and associated components containing radioactive liquids outside containment.

2. The HMB reviews information on the site geology, hydrology, and the parameters and analyses governing liquid waste movement through the soil, i.e., dilution by groundwater, lateral dispersion, transit time, hydraulic gradient, permeability, and effective porosity, based on single failure assumptions.

II. ACCEPTANCE CRITERIA

Tanks and associated components containing radioactive liquids outside containment are acceptable if failure does not result in radionuclide concentrations in excess of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply,* in an unrestricted area, or if special design features are provided to mitigate the effects of postulated failures for systems not meeting these limits.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this plan as may be appropriate for a particular case.

1. The reviewer will evaluate the consequences of postulated single failures for tanks and associated components outside containment that contain contaminated liquids, where the leaked fluid is capable of affecting the nearest existing or known future water supply* in an unrestricted area. The reviewer will select tanks or components for which a failure is assumed for evaluation purposes based on the radionuclide inventory in the components, and the potential for the contaminated liquids entering the groundwater or a potable water supply.

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a. The radionuclide inventory in failed components is based on assuming 80% of the liquid volume in each component and the design basis failed fuel fraction, i.e., 1% of the fuel producing power in a pressurized water reactor (PWR), or, consistent with an offgas release rate of 100 μCi/sec-MWt after 30 minutes delay for a boiling water reactor (BWR). The radionuclide inventory in failed components is calculated based on the techniques and parameters given in NUREG-0016 and NUREG-0017.

b. The reviewer will consider the design features incorporated to mitigate the effect of a postulated failure, e.g., steel liners in building areas housing components. Normally, because of the potential radionuclide inventory, the failed components that are considered are (1) waste collector tanks, (2) evaporator concentrate tanks, (3) phase separator tanks, and (4) spent resin storage tanks. The components selected for evaluation are based on the individual plant design. Selection of the components should be coordinated with the HMB reviewer to assure that the assessment considers access to the hydrosphere and the magnitude of potential dilution and decay for various pathways as well as size and inventory.

Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks. Credit is not allowed for retention by coatings or leakage barriers outside the building foundation.

2. The radionuclide concentrations at the nearest potable water supply are calculated by using the values of hydrological parameters developed by HMB with the guidance in SRP Section 2.4.12 and using the ETSB calculated radionuclide inventory in failed components.

3. Compliance with the acceptance criteria given in subsection II will be determined by ETSB based on the HMB calculation of radionuclide concentrations at the nearest water supply.*

4. The reviewer may elect to use the applicant's evaluation in lieu of an independent calculation. In this case, the applicant's parameters are verified by ETSB and HMB, as appropriate.

IV. EVALUATION FINDINGS

If the review confirms that the consequences of liquid-containing tank failures would be acceptable according to the criteria stated in subsection II, conclusions of the following type are provided for the staff's safety evaluation report:

*"Supply" means a well or surface water intake that is used as a water source for direct human consumption or indirectly through animals, crops or food processing. "Known future" water supply means potential wells or surface water intakes which are identified, or may be reasonably deduced from available information.

"The consequences of tank and associated component failures which could result in contaminated liquid releases to the environs were evaluated for tanks and components containing radioactive materials located outside reactor containment. The scope of the review included the calculation of radionuclide inventories in station components at design basis fission product levels, the mitigating effects of the plant design, and the effect of site geology and hydrology. Radionuclide concentrations were calculated at the nearest potable water supply and at the nearest surface water supply in an unrestricted area.

"The basis for acceptance in our review has been that the postulated failures would not result in radionuclide concentrations in excess of 10 CFR Part 20 limits at the water sources considered above.

"Based on the foregoing evaluation we conclude that the provisions incorporated in the applicant's design to mitigate the effects of tank and component failures involving contaminated liquids are acceptable."

V. REFERENCES

1. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors."

2. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors."

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SECTION 15.7.4 RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

1. The review is concerned with the selection of values of plant parameters for use in calculating the radiological consequences of a fuel handling accident, and the selection of the dose computation model, including assumptions of transport mechanisms and rates from the fuel handling area to the atmosphere, breathing rates, dose conversion factors, and other physical and biological data that may affect the calculated dose.

2. The calculated doses are compared with the appropriate exposure guidelines to determine the acceptability of the nearest exclusion area boundary and low population zone (LPZ) outer boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

The ETSB reviews the fuel handling building ventilation system filters in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

1. The acceptance criteria for those reactor operating parameters and other assumptions and conditions that determine the amounts of fission product activity assumed to be available for release above the fuel pool are largely given in Regulatory Guide 1.25 (Ref. 2). These include such items as gap activities, pool decontamination factors, and radial peaking factors.

2. The dose to an individual from a postulated fuel handling accident should be well within 10 CFR Part 100 exposure guidelines. At the construction permit (CP) review stage, the doses calculated should allow adequate margin for uncertainties to assure that the doses will be well within 10 CFR Part 100 guidelines at the operating license (OL) review stage.

3. In view of the possibility of a fuel handling accident, an ESF filter system is required for the spent fuel storage facility to reduce the potential consequences.

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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 areas 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.

1. The relevant portion of Chapter 15 of the applicant's safety analysis report (SAR) should be read to assure that the values of those fuel parameters which affect fission product release and fuel pool iodine decontamination factors, including the maximum fuel rod pressurization, peak linear power density for the highest power assembly, maximum centerline operating fuel temperature for the peak assembly, average burnup for the peak assembly, and minimum water depth between the top of any damaged fuel rods and the fuel pool surface are such that Regulatory Guide 1.25 (Ref. 2) is applicable. Where Regulatory Guide 1.25 is applicable, Chapter 15 and the technical specifications of the SAR are reviewed to assure that the "conservative analysis" assumptions given for the fuel handling accident agree with those listed in the guide.

   Where the values of the fuel parameters are such that Regulatory Guide 1.25 is not applicable, an individual case-by-case analysis is performed. The AAB Branch Chief is consulted as to appropriate action.

2. Three important parameters affecting the radiological consequences of a fuel handling accident are not covered in Regulatory Guide 1.25. These are the reactor design (stretch) power level, the earliest time after reactor shutdown that fuel handling operations commence, and the number of fuel rods assumed to be damaged in a fuel handling accident. The reactor design power level is obtained from Section 1.1 or Chapter 15 of the applicant's SAR. The "conservative analysis" assumptions listed in Chapter 15 of the SAR are used to determine the earliest time after shutdown for fuel handling and the number of fuel rods damaged. The "conservative analysis" assumptions listed in Chapter 15 are generally considered acceptable and can be used for obtaining the earliest time after shutdown that fuel handling commences (typical values are 24 hours for a boiling water reactor and 100 hours for a pressurized water reactor) and the number of fuel rods damaged. All the pins in one fuel bundle are assumed damaged unless the applicant's analysis shows additional damage, in which case the applicant's assumptions should be adopted. The technical specifications are also reviewed to confirm the earliest time after shutdown that fuel handling commences.

3. The credit to be given for engineered safety features such as filter systems must be determined. For the filters themselves, verification of acceptability and efficiencies is provided by the ETSB in SRP 6.5.1. In a dual containment system where the fuel building may be exhausted through the standby gas-treatment system (SGTS), the AAB must determine the relation between the operational modes of the SGTS and the time sequence of the accident in order to give proper credit.

4. Sections 2.1.2 and 2.1.3 of the SAR are examined to determine the minimum distances to the exclusion area boundary and to the LPZ outer boundary. Following the procedures

given in Standard Review Plan (SRP) 2.1.2 and SRP 2.1.3, the reviewer confirms the validity of the applicant's values. From the SAR, the reviewer also obtains relevant information (e.g., locations and time durations) concerning activities within the exclusion area boundary unrelated to facility operation.

5. The SAB is requested to furnish suitable X/Q values to be used in analyzing the consequences of the accident. X/Q values are obtained not only at the nearest exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion area boundary where there may be significant activities unrelated to facility operation.

6. The relevant plant parameters and the X/Q values obtained from SAB are used as input to an appropriate digital computer code (Ref. 3) and the doses due to a postulated fuel handling accident are calculated at the nearest exclusion area boundary, the outer boundary of the LPZ, and at those locations within the exclusion area boundary where significant activities unrelated to facility operation may occur.

7. The calculated doses are compared to the acceptance criteria in Section II. Where results of the dose calculations indicate the guidelines may be exceeded, alternatives which would reduce the doses to an acceptable level are explored (e.g., increased distance, better filters); the feasibility of the alternatives is also examined. The AAB Branch Chief is to be consulted as to appropriate action.

8. The applicant's SAR is examined to assure that an ESF filter system to mitigate the radiological consequences of a fuel handling accident are included in the design of the spent fuel storage facility.

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 staff has evaluated the applicant's analysis of postulated fuel handling accidents and finds the assumptions and calculational techniques acceptable. After performing an independent analysis of the radiological consequences of a fuel handling accident to any individual located at the nearest exclusion area boundary, at the outer boundary of the low population zone (LPZ), or at any point within the exclusion area boundary where there may be significant activity unrelated to plant operation, the staff concludes that the doses are well within the guideline values of 10 CFR Part 100."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. 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."

3. Computer codes are currently under development. Documentation will be published as a NUREG report.

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SECTION 15.7.5 SPENT FUEL CASK DROP ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary Systems Branch (ASB)
Hydrology-Meteorology Branch (HMB)
Effluent Treatment Systems Branch (ETSB)
Core Performance Branch (CPB)

I. AREAS OF REVIEW

The AAB reviews accidents involving a drop of a spent fuel cask, as described in the applicant's safety analysis report (SAR), Section 15.7.5. The ASB evaluates the spent fuel cask handling system in SRP Section 9.1.4. The points covered in the review are as follows:

1. ASB is consulted to verify the potential drop height during handling of a loaded cask and the procedures for handling the cask with respect to the impact limiter. If the handling procedures meet all applicable criteria, then the radiological consequences of a spent fuel cask drop accident need not be estimated.

2. A design basis radiological analysis is performed if a cask drop exceeding 30 feet can be postulated or if limiting devices are removed during cask handling within the plant so the 30-foot drop height is exceeded. If the radiological consequences of a cask drop accident are to be computed, then information on whether building leaktightness can be expected after a cask drop is obtained from ASB (e.g., whether the technical specifications require large doors to be closed during fuel handling or whether ventilation systems should be operating and whether the building leaktightness would be violated by the cask drop). AAB assumes that only gap activity is released if CPB verifies in SRP Section 9.1.2 that a change in the geometric spacing of the fuel will not occur and verifies in SRP Section 9.1.3 that water in the spent fuel pool will not be drained as a result of a cask drop.

3. The SAR and technical specifications are reviewed and the relevant plant parameters are evaluated for incorporation into the dose computation model. The model incorporates conservative transport mechanisms and rates from the fuel release to the atmosphere, suitable breathing rates, dose conversion factors, and other physical and biological data that may affect the dose. The X/Q data are obtained from HMB. It may be found appropriate to utilize analyses from previous cases to determine the

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consequences on a generic basis. ETSB provides filter efficiencies of systems used to mitigate the consequences of a cask drop.

4. The calculated doses are compared with exposure guidelines to determine the acceptability of the exclusion and low population zone (LPZ) distances and to confirm the adequacy of engineered safety features (ESF) provided for the purpose of mitigating potential doses from spent fuel cask drop accidents.

5. ASB is consulted for verification that a cask drop or tipping will not damage fuel in either the spent fuel storage building (or in the containment building if applicable). If the handling procedures are such that spent fuel can be damaged, an analysis of the resulting offsite doses should be reviewed.

II. ACCEPTANCE CRITERIA

1. The plant design with regard to spent fuel cask drop accidents is acceptable without calculation of radiological consequences if potential cask drop distances are less than 30 feet and appropriate impact limiting devices are employed during cask movements.

2. If the radiological consequences of a spent fuel cask drop accident are to be considered, the plant design is acceptable in this regard if the doses to an individual at the exclusion area boundary and LPZ outer boundary distances are calculated to be well within 10 CFR Part 100 exposure guidelines. At the construction permit (CP) review stage, the doses calculated should allow adequate margin for uncertainties to assure that the doses will be well within 10 CFR Part 100 guidelines at the operating license (OL) review stage.

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 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.

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 first step in the review procedure is to determine, with the assistance of the ASB as described in Subsection I, whether radiological consequences of a spent fuel cask drop accident need be evaluated. If a radiological consequence calculation is found to be necessary, the procedure is as follows:

1. The fuel element gap inventory is determined in a manner similar to that for a fuel handling accident (see Ref. 2). The differences are that a longer decay time is allowed (earliest time after reactor fueling that cask loading operations commence) and the number of fuel elements involved is based on the largest capacity cask available or projected to be available.

2. If the drop is assumed to occur inside the refueling facility at a time when the facility is closed, at a minimum negative pressure of 0.25-inch water gauge, and ESF-grade charcoal filtration is available, credit may be allowed for iodine filtration. For the filters themselves, verification of acceptability and efficiencies is provided by the [illegible]SB. In a dual containment design where the fuel building may be exhausted through the standby gas treatment system (SGTS), AAB determines the relationship of the operational modes of the SGTS to the time sequence of the accident in order to give proper credit.

3. If the spent fuel cask drop is assumed to occur at a time when the facility is open to the outside atmosphere, an untreated puff release is assumed.

4. If a spent fuel cask is utilized in a containment structu[illegible]ich is not isolated during fuel cask transfer and ASB has determined that cask drop or tipping on spent fuel can occur, the radiological doses from all failed fuel should be estimated.

5. The HMB is requested to furnish suitable X/Q values to analyze the consequences of the accident. X/Q values are obtained not only at the exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion area boundary where there may be significant activities unrelated to plant operation.

6. The relevant plant parameters and the X/Q values obtained from the HMB are used to compute doses. The doses due to a postulated spent fuel cask drop accident are calculated at the exclusion area boundary, the outer boundary of the LPZ, and at those locations within the exclusion area boundary where there may be significant activities unrelated to plant operation.

7. The calculated doses are compared with the acceptance criteria in Section II. Where results of the dose calculations indicate the guidelines may be exceeded, alternatives which would reduce the dose to acceptable levels are explored (e.g., increased decay time, better filters); the feasibility of the alternatives is also examined. The AAB Branch Chief is to be consulted as to appropriate action.

IV. EVALUATION FINDINGS

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

"We have evaluated the applicant's analysis of postulated spent fuel cask drop accidents and find the assumptions and calculational techniques acceptable. After performing an independent analysis of the radiological consequences to any individual located at the exclusion area boundary, at the outer boundary of the low population zone (LPZ), or at any point within the exclusion area boundary where there may be significant activities unrelated to plant operation, we conclude that the doses are well within the guideline values of 10 CFR Part 100. The doses are listed in Table ____. "

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. 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," and SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents."

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SECTION 15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Anticipated transients are transients expected to occur during the life of the plant. Anticipated transients without scram (ATWS) are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram). The failure of the reactor to shut down quickly during these transients can lead to unacceptable reactor coolant system pressures and to fuel damage. Typical transients that may have unacceptable consequences if there is a scram failure are: loss of feedwater, loss of load, turbine trip, inadvertent control rod withdrawal, loss of a-c power, loss of condenser vacuum, and, in a boiling water reactor (BWR), closure of main steamline isolation valves. The effects of all ATWS events must be evaluated, including a complete analysis for those resulting in the most severe consequences. Each should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1).

According to the licensing position on ATWS (Ref. 2), plants for which a construction permit (CP) application is filed after October 1, 1976 must have reactor shutdown systems of such nigh reliability that ATWS events need not be considered. This standard review plan (SRP), therefore, applies only to plants where application was made for a CP before that date.

For such plants, the review of ATWS events is concerned with the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted results and consequences of the transient.

The sequence of events described in the applicant's SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the effect of the reactor protection system, engineered safety systems, other operational and control systems, and operator action on the course of the transient and the securing and maintenance of the reactor in a safe condition. The

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EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and operational safeguards controls and instrumentation systems (other than the scram system which is assumed to have failed) will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

CSB reviews ATWS effects on the containment system with regard to the effects of the resulting pressures and temperatures on its functional capabilities.

MEB reviews the effects of ATWS events on the structural integrity of the reactor coolant pressure boundary.

AAB evaluates the fission product release assumptions used in determining any offsite releases and verifies that the radiological consequences resulting from ATWS events are within acceptable limits. (See attached appendix.)

The predicted results of ATWS events are reviewed to assure that the consequences meet the acceptance criteria given in Section II below. Further, the results of the analyses are reviewed to ascertain that the values of the pertinent initial conditions and system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The general objectives in the review of ATWS events are:

   a. To identify the ATWS events that are most limiting.

   b. To verify that for the most limiting events the specific criteria applicable to ATWS events are met.

2. The specific criteria that apply depend on the date the CP application is filed and whether the need to take ATWS into account was noted in the staff's safety evaluation report (SER) or in the report of the Advisory Committee on Reactor Safeguards (ACRS). If the need to take ATWS into account was noted in either of these reports and if an application for a CP is filed before October 1, 1976, then the following criteria (Appendix A of Ref. 2) apply:

a. Calculation of Consequences

The calculated radiological consequences should be within the guideline values set forth in 10 CFR Part 100. In addition, the limits listed below on calculated system pressure, fuel performance, and containment conditions should be met.

1. Reactor Coolant System Pressure

The maximum acceptable calculated transient reactor coolant system pressure should be based on the system boundary pressure limit or the fuel pressure limit, whichever is more restrictive:

(i) Reactor Coolant System Boundary Pressure Limit

The calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in the ASME Code, Section III (Ref. 3).

(ii) Fuel Pressure Limit

The calculated reactor coolant system transient pressure should not exceed a value for which tests and analyses demonstrate that there is no significant safety problem with the fuel.

2. Fuel Thermal and Hydraulic Performance

(i) The calculated average enthalpy of the hottest fuel pellet should not result in significant cladding degradation or significant fuel melting.

(ii) A calculated critical heat flux event should not occur unless the calculated peak cladding temperature can be shown not to result in significant cladding degradation.

3. Containment Conditions

Calculated maximum containment pressure should not exceed the design pressure of the containment structure. Equipment located within the containment that is relied upon to mitigate the consequences of ATWS should be qualified by testing in the combined pressure, temperature, and humidity environment conservatively predicted to occur during the course of the event.

b. Evaluation Techniques

Analysis models and techniques, including computer codes, used for conservative evaluations of the consequences of postulated ATWS events, together with associated assumptions and parameters, should be described and justified in topical reports.

c. Review of Reactor Shutdown System Design

A review of the reactor shutdown system design should be made with the aim of identifying and correcting areas that might be particularly vulnerable to common mode failures.

d. Diversity Requirements

Design changes to make the calculated consequences of postulated ATWS acceptable should not rely on equipment which has a failure mode common with the anticipated transient or the shutdown system. To the extent practical, the equipment involved in the design should operate on a different principle from equipment in the shutdown system. As a minimum, the equipment relied on to make the consequences of a postulated ATWS event acceptable should not include equipment identical to equipment in the associated shutdown system. Such designs should be shown not to result in violations of safety criteria for steady-state, transient, or accident conditions and should not adversely affect the operation of any safety-related systems.

III. REVIEW PROCEDURES

The review procedures below are used during both the CP and operating license (OL) reviews for plants for which this plan is applicable. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL stage, final values should be used in the analysis and the reviewer should compare these to the expected values, control system bands, and limiting safety system settings included in the proposed technical specifications.

The description of the ATWS events (Table 15-1 of Ref. 1) presented by the applicant in the SAR are reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which engineered safety system actuation instrumentation and controls are required to function, and the credit taken for reactor trip alarms to inform the operator of abnormal conditions.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

The applicant should present a quantitative analysis in the SAR of the ATWS events that are determined to be most limiting. The justification for selecting these events should be given. The RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of this protection, engineered safety, and other systems needed to limit the consequences of the events to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes,

interlocks, and the feasibility of manual operation where the SAR states that operator action is needed or expected. He also verifies that the applicant has reviewed the design of the reactor shutdown system and identified and corrected areas which are vulnerable to common mode failures. Both the RSB and EICSB reviewers verify that any design changes made to mitigate the consequences of the ATWS events have no failure mode common with the cause of the anticipated transient or with the shutdown system.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam system are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients used by the applicant in his analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding maximum pressure in the reactor coolant and main steam systems and fuel performance limits. For each ATWS event the variation with time during the transient of the core and barrier performance parameters listed in the "Event Evaluation" section of Chapter 15 of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for ATWS events are compared to those predicted for other similar plants to verify that they are within the expected range.

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:

> "A number of plant transients can be affected by a failure of the scram system to function. For a PWR* the most important include loss of feedwater, loss of load, inadvertent control rod withdrawal, and loss of a-c power. All postulated anticipated transients which can be affected by a failure to scram have been reviewed.** It was found that the most limiting ATWS events, with respect to possible fuel damage and

*For a BWR they are loss of condenser vacuum, closure of the main steam isolation valves, and turbine trip.

**The SER should present one statement for all similar transients.

pressure within the reactor coolant and main steam systems, were the _____ and _____events respectively. The events were evaluated by the applicant using a mathematical model that has been reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis showed a peak reactor coolant system pressure of _____ psia which is within the 'emergency conditions' defined in Section III of the ASME Boiler and Pressure Vessel Code. The calculated (fuel clad temperature, DNBR, CPR, or MCHFR) following a postulated _____ transient without scram is _____ and will not result in significant fuel damage. The radioactivity release from this event has been calculated and the resulting doses found to be within the guidelines of 10 CFR Part 100.

"Based on the above evaluation, the staff concludes that the plant design is acceptable with regard to anticipated transients without scram."

V. REFERENCES

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

2. Regulatory Staff, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, U. S. Atomic Energy Commission, Sept. 1973.

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-3000, "Design," American Society of Mechanical Engineers.

APPENDIX

STANDARD REVIEW PLAN 15.8

RADIOLOGICAL CONSEQUENCES OF AN ATWS EVENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The review of ATWS events will include estimates of the radiological consequences. These estimated consequences will be compared to those estimated for design basis events as well as the exposure guidelines of 10 CFR Part 100 and Regulatory Guides 1.3 and 1.4. If the consequence estimates for any ATWS event are significant when compared to those for design basis events or require imposition of a technical specification to limit the consequences, then that ATWS event will be included as a design basis event in the staff safety evaluation. The review method and plan are under development and will rely on the result of current staff reviews of vendor analyses of ATWS events.

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

SECTION 16.0 TECHNICAL SPECIFICATIONS

REVIEW RESPONSIBILITIES

Primary - Licensing Project Manager and Standard Technical Specification Group

Secondary - Review Branches

Section 50.36 of 10 CFR of Part 50, requires that each operating license issued by the Commission contain technical specifications that set forth the limits, operating conditions, and other requirements imposed upon facility operation for the protection of the health and safety of the public. As part of the regulatory standardization effort, the staff has prepared generic standard technical specifications (STS) for each of the light water reactor nuclear steam supply system and associated balance of plant equipment systems. These STS's are subject to revision and the latest versions are available from the Office of Nuclear Reactor Regulation, NRC. The initial implementation of the STS program was made on the D. C. Cook operating license issued in October 1974. All subsequent operating licenses issued by the Commission will utilize the appropriate generic STS as the basis for issuance of Appendix A of licenses, "Technical Specifications."

Applicants should use the current generic STS as the basis for preparation of proposed Appendix A "Radiological Technical Specifications" for Section 16.0 of preliminary safety analysis reports (PSAR). The proposed Appendix A "Technical Specifications" will be reviewed to determine that the content and format are consistent with the applicable generic STS. Special attention will be given to those specifications which deviate from the generic STS to determine that proposed differences are justified on the basis of uniqueness in plant design or other considerations. Specifications so identified will be reviewed in detail to identify areas that may influence the acceptability of the final facility design. In particular, this portion of the review will determine the acceptability of proposed specifications that describe features affecting the type, capacity, number or performance of surveillance activities involving safety-related systems.

Numerical values, graphs and other data proposed will not be as complete as specified in the generic STS because of the preliminary nature of the plant design. The review of information that is provided in this area will be limited to determining that the values are in reasonable agreement with the expected operational capability of the plant.

Applicants should use the current generic STS as the bases for preparation of proposed Appendix A "Radiological Technical Specifications" for Section 16.0 of final safety analysis reports (FSAR). The Appendix A technical specifications submitted in support of an operating license will be the finalized version of those specifications originally included in the PSAR and will reflect the final refinements in design, results of tests and expected method of operation.

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 review of these specifications will be done on an item by item basis to determine the comparability with the current generic STS. This comparative review will identify in a detailed manner all deviations from the generic standard in applicability, format and specific content. The noted specification differences and supporting bases will be reviewed to determine their acceptability on the basis of plant design differences or other considerations. Deviations that are found to be acceptable at this stage of review will be evaluated for suitability of incorporation into one or more of the generic STS's.

The numerical values, graphs, tables and other data proposed for each specification should be as complete as specified in the generic STS. This information will be reviewed to insure conformance with material presented in applicable portions of the FSAR as summarized in the supporting basis for each specification.

Each generic STS will be maintained current and updated periodically to reflect:

(1) changes in classes of plants or modifications of nuclear steam supply systems or balance of plant equipment systems.

(2) Revised regulatory requirements.

(3) Experience obtained by the NRC staff in reviewing proposed technical specification changes from licensees.

(4) Operational experience obtained from licensees and Office of Inspection and Enforcement inspection personnel.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Section 17.1 QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION PHASES

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

QAB reviews and evaluates the description of the quality assurance (QA) program for the design and construction phases in each application for a construction permit (CP), a manufacturing license, or a standardized design approval in accordance with applicable portions of this section of the Standard Review Plan.

Pre-Docketing

Prior to docketing a CP application, the NRC performs a substantive review of the applicant's QA program description relative to ongoing design and procurement activities. This review and associated inspection are performed immediately after tendering of a CP application to determine that a satisfactory QA program has been established and is being implemented.

The pre-docketing substantive review places particular emphasis on the areas of organization, QA program, design control, procurement document control, and audit. The application is not docketed unless the established and implemented program in these areas has no substantive deviation from NRC QA guidance applicable to activities conducted prior to docketing. Representatives from the offices of NRR and IE may meet with the applicant's representatives nine to twelve months prior to tendering of the application to provide a clear understanding of what is expected in the QA program description and the implemented program in order for the program to be accepted during the substantive review and associated inspection.

Where an NRC-accepted QA topical report is referenced in the application, the referenced QA program is not re-reviewed except for conformance to the applicable Regulatory Guides in effect at the time of tendering the application. For the case of CP applications referencing a standard design that includes an approved QA program directly or by reference, the applicant need not conform to new Regulatory Guides unless they contain regulatory positions determined to be significant to safety.

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.

Post-Docketing

The QAB review, after docketing, covers the QA controls to be applied by the applicant and principal contractors to activities that may affect the quality of structures, systems, and components important to safety. These activities include site testing and evaluation (starting with evaluation of exposed excavated surfaces, soil compaction, and testing), designing, purchasing, fabricating, constructing, handling, shipping, storing, cleaning, erecting, installing, inspecting, and testing. This review extends to the determination of how the applicable requirements of the eighteen criteria of Appendix B to 10 CFR 50 are satisfied by the proposed QA program.

The areas of review are as follows:

1. ORGANIZATION

   A. Organizational description and charts of the lines, interrelationships and areas of responsibility and authority for all organizations performing quality-related activities, including the applicant's organization and principal contractors (architect engineer, nuclear steam supply system vendor, constructor, and construction manager when other than the constructor).

   B. Organizational location, degree of independence from the performing organization, and authority of the individuals assigned the responsibility for performing QA functions.

   C. Organizational provisions for assuring the proper implementation of the QA program.

2. QUALITY ASSURANCE PROGRAM

   A. Scope of the QA program.

   B. Provisions to assure proper definition of the QA program.

   C. Programmatic provisions to assure proper implementation of the QA program.

   D. Provisions to assure adequacy of personnel qualifications.

3. DESIGN CONTROL

   A. Scope of the QA program for design activities.

   B. The organizational structure, activity, and responsibility of the positions or groups responsible for design activities.

C. Provisions to carry out design activities in a planned, controlled, and orderly manner.

D. Provisions for interface control.

E. Provisions to verify or check the technical adequacy of design documents.

F. Provisions to control design changes.

4. PROCUREMENT DOCUMENT CONTROL

A. Provisions which assure that applicable regulatory requirements, technical requirements, and QA program requirements are included or referenced in procurement documents.

B. Provisions for review and approval of procurement documents.

5. INSTRUCTIONS, PROCEDURES, AND DRAWINGS

A. Provisions for assuring that activities affecting quality are prescribed by and accomplished in accordance with documented instructions, procedures, or drawings.

B. Provisions for including quantitative and qualitative acceptance criteria in instructions, procedures, and drawings.

6. DOCUMENT CONTROL

A. Provisions to assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed.

B. Provisions to prevent the inadvertent use of obsolete or superseded documents.

7. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

A. Provisions for the control of purchased material, equipment, and services; for selection of suppliers; and for assessing the adequacy of quality.

B. Provisions to assure that documented evidence of the conformance of material and equipment to procurement requirements is available at the plant site prior to installation or use.

8. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

A. Provisions to identify and control materials, parts, and components.

B. Provisions to assure that incorrect or defective items are not used.

9. CONTROL OF SPECIAL PROCESSES

A. Provisions to assure the acceptability of special processes such as welding, heat treating, nondestructive testing, and chemical cleaning.

B. Provisions to assure that special processes are performed by qualified personnel using qualified procedures and equipment.

10. INSPECTION

A. Provisions for the inspection of activities affecting quality, including the items and activities to be covered.

B. Organizational responsibilities and qualifications established for individuals or groups performing inspections.

C. Prerequisites to be provided in the written inspection procedures with provisions for documenting and evaluating inspection results.

11. TEST CONTROL

A. Provisions for tests which assure that structures, systems, and components will perform satisfactorily in service.

B. Prerequisites to be provided in written test procedures with provisions for documenting and evaluating test results.

C. Personnel qualification programs established for test personnel.

12. CONTROL OF MEASURING AND TEST EQUIPMENT

Provisions to assure that tools, gages, instruments, and other measuring and testing devices are properly identified, controlled, calibrated, and adjusted at specified intervals.

13. HANDLING, STORAGE, AND SHIPPING

Provisions to control handling, storage, shipping, cleaning, and preservation of items in accordance with work and inspection instructions to prevent damage, loss, and deterioration by environmental conditions such as temperature or humidity.

14. INSPECTION, TEST, AND OPERATING STATUS

Provisions to indicate the inspection, test, and operating status of items to prevent inadvertent use or bypassing of inspection and tests.

15. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Provisions to control the use or disposition of nonconforming materials, parts, or components.

16. CORRECTIVE ACTION

Provisions to assure that conditions adverse to quality are promptly identified and corrected and that measures are taken to preclude repetition.

17. QUALITY ASSURANCE RECORDS

Provisions for the identification, retention, retrieval, and maintenance of records that furnish evidence of activities affecting quality.

18. AUDITS

A. Provisions for audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the QA program.

B. Responsibilities and procedures for auditing, documenting and reviewing audit results, and designating management levels to review and assess audit results.

II. ACCEPTANCE CRITERIA

The applicant (and its principal contractors such as the NSSS vendor, A/E, constructor and construction manager) must establish a QA program for the design and construction phases in accordance with Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The applicant's QA program (including its principal contractors) must describe in the PSAR or SSAR how each criterion of Appendix B will be met. The acceptance criteria used by the QAB to evaluate this QA program are listed in the following eighteen subsections. The acceptance criteria include a commitment to comply with the regulatory positions presented in the appropriate issue of the Regulatory Guides including the requirements of ANSI Standard N45.2.12 and Branch Technical Position listed in subsection V. Thus, the commitment constitutes an integral part of the QA program description and requirements. Exceptions and alternatives to these acceptance criteria may be adopted by applicants provided adequate justification is given; the QAB review allows for considerable flexibility in defining methods and controls while still satisfying pertinent regulations. When the QA program description meets the applicable acceptance criteria of this subsection or provides acceptable exceptions or alternatives, the program is considered to be in compliance with pertinent NRC regulations.

The review will ascertain that the commitments and the description of how the commitments are implemented, to the extent necessary, are objective and stated in inspectable terms.

The Organization (17.1.1) elements responsible for the QA program are acceptable if:

1A1.* The responsibility for the overall program is retained and exercised by the applicant.

1A2. The applicant has identified and described major delegation of work involved in establishing and implementing the QA program or any part thereof to other organizations.

1A3. When major portions of the applicant's program are delegated:

a. Applicant describes how responsibility is exercised for the overall program.

b. Applicant evaluates the performance (frequency and method stated - once per year although longer cycle acceptable with other evaluations of individual elements) of work by the delegated organization.

c. Qualified individual(s) or organizational element(s) are identified within the applicant's organization as responsible for the quality of the delegated work prior to initiation of activities.

1A4. Clear management controls and effective lines of communication exist for QA activities among the applicant and the principal contractors to assure direction of the QA program.

1A5. Organization charts identify the "onsite" and "offsite" organizational elements which function under the cognizance of the QA program (such as design engineering, procurement, manufacturing, construction, inspection, test, instrumentation and control, nuclear engineering, etc.) and the lines of responsibility.

1A6. The applicant (and principal contractors) describes the QA responsibilities of each of the organizational elements noted on the organization charts.

1B1. The applicant (and principal contractors) identifies a position that retains overall authority and responsibility for the QA program (normally, this position is QA Manager) and this position has the following characteristics:

a. Is at the same or higher organization level as the highest line manager directly responsible for performing activities affecting quality (such as

*The alphanumeric designation for each acceptance criterion in subsection II indicates its relationship to the areas of review identified in subsection I.

engineering, procurement, construction, and operation) and is sufficiently independent from cost and schedule.

b. Has effective communication channels with other senior management positions.

c. Has responsibility for approval of QA Manual(s).

d. Has no other duties or responsibilities unrelated to QA that would prevent his full attention to QA matters.

1B2. Verification of conformance to established requirements is accomplished by individuals or groups who do not have direct responsibility for performing the work being verified.

1B3. Persons and organizations performing QA functions have direct access to management levels which will assure the ability to:

a. Identify quality problems.

b. Initiate, recommend, or provide solutions through designated channels.

c. Verify implementation of solutions.

1B4. Designated QA personnel, sufficiently free from direct pressures for cost/schedule, have the responsibility delineated in writing to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material.

1C1. Policies regarding the implementation of the QA program are documented and made mandatory. These policies are established at the Corporate President or Vice President level.

1C2. Position description (see 1B1) assures that the individual directly responsible for the definition, direction, and effectiveness of the overall QA program has sufficient authority to effectively implement responsibilities. This position is to be sufficiently free from cost and schedule responsibilities. Qualification requirements for this individual are established in a position description which includes the following prerequisites:

a. Management experience through assignments to responsible positions.

b. Knowledge of QA regulations, policies, practices, and standards.

c. Experience working in QA or related activity in reactor design, construction, or operation or in a similar high technological industry.

The qualifications of the QA Manager should be at least equivalent to those described in Section 4.4.5 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," as endorsed by the regulatory positions in Regulatory Guide 1.8.

1C3. The person at the construction site responsible for directing and managing the site QA program has appropriate organizational position, responsibilities, and authority to exercise proper control over the QA program. This individual is free from non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented.

Activities related to Quality Assurance Program (17.1.2) are acceptable if:

2A1. The scope of the QA program includes:

a. A commitment that activities affecting structures, systems, and components important to safety will be subject to the applicable controls of the QA program.

b. A commitment that the preoperational test program will be conducted in accordance with the QA program and a description of how the QA program will be applied.

c. A commitment that the development, control, and use of computer code programs will be conducted in accordance with the QA program and a description of how the QA program will be applied.

d. The identification of structures, systems, and components and related consumables covered by the QA program and controlled measures identifying authorized personnel to approve changes to this list and describing methods controlling its distribution.

e. The identification of fire protection in Section 9.5.1 as a system covered by the QA program or identification of the QA controls for fire protection. These controls are reviewed and accepted using the guidelines contained in BTP ASB-9.5.1.

f. A commitment that special equipment, environmental conditions, skills, or processes will be provided as necessary.

2A2. A brief summary of the company's corporate QA policies is given.

2B1. Provisions are established to assure that procedures required to implement the QA program are consistent with QA program commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official.

a. The organizational group or individual having responsibility for the policy statement should be identified.

b. The procedural controls of the principal contractors should be provided for the applicant's review with documented agreement of acceptance prior to initiation of activities affected by the program.

2B2. Provisions are included for notifying NRC of changes (1) for review and acceptance in the accepted description of the QA program as presented or referenced in the SAR or SSAR prior to implementation, and (2) in organizational elements within 30 days after announcement. (Note - editorial changes or personnel reassignments of a non-substantive nature do not require NRC notification).

2B3. The applicant (and the principal contractors) commits to comply with the regulatory position in the appropriate issue of the Regulatory Guides; with 10 CFR 50.55a; and with the requirements of the endorsed ANSI Standard N45.2.12 listed in subsection V. For systems, components, and structures covered by the ASME Code Section III (Classes 1, 2 and 3), the quality assurance code requirements should be supplemented by the specific guidance addressed in the regulatory positions of the applicable Regulatory Guides. The commitment identifies the Regulatory Guides and ANSI standard by number, title, and revision or date. Any alternatives or exceptions are clearly identified and supporting information presented in the docket. QA Regulatory Guides should be addressed which have an implementation date prior to the submittal or docket date of the QA program description.

Although primary responsibility for Regulatory Guides 1.26 and 1.29 is assigned to ASB (SRP Sections 3.2.1 and 3.2.2), their use as acceptance criteria in this SRP section is necessary to assure that adequate quality assurance requirements are specified for systems, components, and structures addressed by those guides.

2B4. Existing or proposed QA procedures are identified reflecting that each criterion of 10 CFR 50 Appendix B will be met by documented procedures.

2B5. A description is provided that emphasizes how the docketed QA program description, particularly the Regulatory Guides and the endorsed ANSI standard listed in subsection V, will be properly carried out.

2C1. A description is provided of how management (above or outside the QA organization) regularly assesses the scope, status, adequacy, and compliance of the QA program to 10 CFR 50 Appendix B. These measures should include:

a. Frequent contact with program status through reports, meetings, and/or audits.

b. Performance of an annual assessment preplanned and documented. Corrective action is identified and tracked.

2C2. Quality-related activities (such as design, procurement, and site investigation) initiated prior to formal NRC acceptance of the QA program are controlled under a QA program in accordance with this SRP and, accordingly, with the requirements of 10 CFR 50 Appendix B. Approved procedures and trained personnel should be available to implement the applicable portion of the QA program prior to the initiation of the activity.

2C3. A summary description is provided on how responsibilities and control of quality-related activities are transferred from the principal contractors to the applicant during the phaseout of design and construction and during preoperational testing and plant turnover.

2D. An indoctrination and training program is established such that:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

b. Personnel verifying activities affecting quality are trained in the principles, techniques, and requirements of the activity being performed.

c. For formal training programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, reexamining, and/or recertifying as determined by management or program commitment.

e. The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58.

Activities related to Design Control (17.1.3) are acceptable if:

3A. The scope of the design control program includes design activities associated with the preparation and review of design documents including the correct translation of applicable regulatory requirements and design bases into design, procurement and procedural documents. Included in the scope are such activities as field design engineering; physics, seismic, stress, thermal, hydraulic, radiation, and the SAR accident analyses; associated computer programs; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and quality standards.

3B. Organizational responsibilities are described for preparing, reviewing, approving, and verifying design documents such as system descriptions, design input and criteria, design drawings, design analyses, computer programs, specifications, and procedures.

3C1. Errors and deficiencies in approved design documents, including design methods (such as computer codes), that could adversely affect structures, systems, and components important to safety are documented; and action is taken to assure that all errors and deficiencies are corrected.

3C2. Deviations from specified quality standards are identified and procedures are established to ensure their control.

3D. Internal and external design interface controls, procedures, and lines of communication among participating design organizations and across technical disciplines are established and described for the review, approval, release, distribution, and revision of documents involving design interfaces.

3E1. Guidelines or criteria are established and described for determining the method of design verification (design review, alternate calculations, or test).

3E2. Procedures are established and described for design verification activities which assure the following:

a. The verifier is qualified and is not directly responsible for the design (i.e., neither the performer or his immediate supervisor). In exceptional circumstances, the designer's immediate supervisor can perform the verification provided:

(1) The supervisor is the only technically qualified individual.

(2) The need is individually documented and approved in advance by the supervisor's management.

(3) QA audits cover frequency and effectiveness of use of supervisors as design verifiers to guard against abuse.

b. Design verification, if other than by qualification testing of a prototype or lead production unit, is completed prior to release for procurement, manufacturing, construction or to another organization for use in other design activities. In those cases where this timing cannot be met, the design verification may be deferred, providing that the justification for this action is documented and the unverified portion of the design output document and all design output documents, based on the unverified data, are appropriately identified and controlled. Construction site activities associated with a design or design change should not proceed without verification past the point where the installation would become irreversible (i.e., require extensive demolition and rework). In all cases, the design verification should be complete prior to fuel load for a plant under construction, or in the case of an operating plant, prior to relying upon the component, system, or structure to perform its function.

c. Procedural control is established for design documents that reflect the commitments of the SAR; this control differentiates between documents that receive formal design verification by interdisciplinary or multi-organizational teams and those which can be reviewed by a single individual (a signature and date is acceptable documentation for personnel certification). Design documents subject to procedural control include, but are not limited to, specifications, calculations, computer programs, system descriptions, SAR when used as a design document, and drawings including flow diagrams, piping and instrument diagrams, control logic diagrams, electrical single line diagrams, structural systems for major facilities, site arrangements, and equipment locations. Specialized reviews should be used when uniqueness or special design considerations warrant.

d. The responsibilities of the verifier, the areas and features to be verified, the pertinent considerations to be verified, and the extent of documentation are identified in procedures.

3E3. The following provisions are included if the verification method is only by test:

a. Procedures provide criteria that specify when verification should be by test.

b. Prototype, component or feature testing is performed as early as possible prior to installation of plant equipment, or prior to the point when the installation would become irreversible.

c. Verification by test is performed under conditions that simulate the most-adverse design conditions as determined by analysis.

3E4. Procedures are established to assure that verified computer codes are certified for use and that their use is specified.

3F1. Design and specification changes, including fields changes, are subject to the same design controls that were applicable to the original design.

3F2. The description of the design control provisions satisfies the criteria of Regulatory Guide 1.64.

Activities related to Procurement Document Control (17.1.4) are acceptable if:

4A1. Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. To the extent necessary, procurement documents should require contractors and subcontractors to provide an acceptable quality assurance program. The review

and concurrence of the adequacy of quality requirements stated in procurement documents is performed by personnel qualified in QA practices.

4A2. Procedures are established to assure that procurement documents identify applicable regulatory, technical, administrative, and reporting requirements; drawings; specifications; codes and industrial standards; test and inspection requirements; and special process instructions that must be complied with by suppliers.

4B1. Organizational responsibilities are described for (1) procurement planning; (2) the preparation, review, approval, and control of procurement documents; (3) supplier selection; (4) bid evaluations; and (5) review and concurrence of supplier QA programs prior to initiation of activities affected by the program. The involvement of the QA organization is described.

4B2. The description of the procurement document control provisions listed above satisfies the regulatory position in Regulatory Guide 1.123.

Activities related to Instructions, Procedures, and Drawings (17.1.5) are acceptable if:

5A. Organizational responsibilities are described for assuring that activities affecting quality are (1) prescribed by documented instructions, procedures, and drawings and (2) accomplished through implementation of these documents.

5B. Procedures are established to assure that instructions, procedures, and drawings include quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples) acceptance criteria for determining that important activities have been satisfactorily accomplished.

Activities related to Document Control (17.1.6) are acceptable if:

6A1. The scope of the document control program is described, and the types of controlled documents are identified. As a minimum, controlled documents include:

a. Design documents (e.g., calculations, drawings, specifications, analyses) including documents related to computer codes.

b. Procurement documents.

c. Instructions and procedures for such activities as fabrication, construction, modification, installation, test, and inspection.

d. As-built drawings.

e. Quality assurance and quality control manuals.

f. Topical reports.

g. SAR.

h. Nonconformance reports.

6A2. Procedures for the review, approval, and issuance of documents and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. The QA organization, or an individual other than the person who generated the document but qualified in quality assurance, reviews and concurs with these documents with regards to QA-related aspects.

6A3. Procedures are established to assure that changes to documents are reviewed and approved by the same organizations that performed the initial review and approval or by other qualified responsible organizations delegated by the applicant.

6A4. Procedures are established to assure that documents are available at the location where the activity will be performed prior to commencing the work.

6B1. Procedures are established and described to assure that obsolete or superseded documents are removed and replaced by applicable revisions in work areas in a timely manner.

6B2. A master list or equivalent document control system is established to identify the current revision of instructions, procedures, specifications, drawings, and procurement documents. When such a list is used, it should be updated and distributed to predetermined responsible personnel.

Activities related to Control of Purchased Material, Equipment, and Services (17.1.7) are acceptable if:

7A1. Organizational responsibilities are described for the control of purchased material, equipment, and services including interfaces between design, procurement, and QA organizations.

7A2. Verification of suppliers' activities during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed, as required, in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

7A3. Selection of suppliers is documented and filed. If an LCVIP letter of confirmation or the "CASE" Register is used to establish the qualifications of the supplier, the documentation should identify the "letter" or "audit" used.

7A4. Procurement of spare or replacement parts for structures, systems, and components important to safety is subject to present QA program controls, to codes and standards, and to technical requirements equal to or better than the original technical requirements, or as required to preclude repetition of defects.

7B1. Receiving inspection is performed to assure:

a. The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

c. Specified inspection, test and other records, (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the nuclear power plant prior to installation or use.

7B2. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

7B3. The supplier furnishes the following records to the purchaser:

a. Documentation that identifies the purchased item and the specific procurement requirements (e.g., codes, standards, and specifications) met by the item.

b. Documentation identifying any procurement requirements that have not been met.

c. A description of those nonconformances from the procurement requirements dispositioned "accept as is" or "repair."

The review and acceptance of these documents should be described in the purchaser's QA program.

7B4. Suppliers' certificates of conformance are periodically evaluated by audits, independent inspections or tests to assure they are valid.

7B5. The description of the control of procurement provisions listed above satisifies the regulatory position in Regulatory Guide 1.38 and Regulatory Guide 1.123.

Activities related to Identification and Control of Materials, Parts, and Components (17.1.8) are acceptable if:

8A. Controls are established and described to identify and control materials (including consumables), parts, and components including partially fabricated subassemblies. The description should include organizational responsibilities.

8B1. Procedures are established which assure that identification is maintained either on the item or on records traceable to the item to preclude use of incorrect or defective items.

8B2. Identification of materials and parts important to the function of structures, systems, and components important to safety can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical mill test reports.

8B3. Correct identification of material, parts, and components is verified and documented prior to release for fabrication, assembling, shipping, and installation.

Activities related to Control of Special Processes (17.1.9) are acceptable if:

9A1. The criteria for determining those processes that are controlled as special processes are described. As complete a listing as possible of special processes, which are generally those processes where direct inspection is impossible or disadvantageous, should be provided. Some examples are welding, heat treating, NDT, and chemical cleaning.

9A2. Organizational responsibilities are described for qualification of special processes, equipment, and personnel.

9B1. Procedures, equipment, and personnel associated with special processes are qualified and are in conformance with applicable codes, standards, QA procedures, and specifications.

9B2. Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel.

9B3. Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

Activities related to Inspection (17.1.10) are acceptable if:

10A. The scope of the inspection program is described that indicates an effective inspection program has been established. Program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed.

10B1. Organizational responsibilities for inspection are described. Individuals performing inspections are other than those who performed or directly supervised the activity being inspected and do not report directly to the immediate supervisors who are responsible for the activity being inspected. If the individuals performing inspections are not part of the QA organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure such as cost and schedule should be reviewed and found acceptable by the QA organization prior to the initiation of the activity.

10B2. A qualification program for inspectors (including NDT personnel) is established and documented, and the qualifications and certifications of inspectors are kept current.

10C1. Inspection procedures, instructions, or checklists provide, as required, for the following:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of item 10B1.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy requirements

10C2. Procedures are established and described to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

10C3. Inspection results are documented, evaluated and their acceptability determined by a responsible individual or group.

Activities related to Test Control (17.1.11) are acceptable if:

11A1. The description of the scope of the test control program indicates an effective test program has been established for tests including proof tests prior to installation and preoperational tests. Program procedures provide criteria for determining the accuracy requirements of test equipment and criteria for determining when a test is required or how and when testing activities are performed.

11B1. Test procedures or instructions provide as required for the following:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

11C1. Test results are documented, evaluated, and their acceptability determined by a responsible individual or group.

Activities related to Control of Measuring and Test Equipment (17.1.12) are acceptable if:

12.1 The scope of the program for the control of measuring and test equipment is described and the types of equipment to be controlled are established. This information indicates an effective calibration program has been established.

12.2 Organizational responsibilities are described for establishing, implementing, and assuring effectiveness of the calibration program

12.3 Procedures are established and described for calibration (technique and frequency), maintenance, and control of the measuring and test equipment (instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment) that is used in the measurement, inspection, and monitoring of structures, systems, and components.

12.4 Measuring and test equipment is identified and traceable to the calibration test data.

12.5 Measuring and test equipment is labeled or tagged or "otherwise controlled" to indicate due date of the next calibration. The method of "otherwise controlled" should be described.

12.6 Measuring and test equipment is calibrated at specified intervals based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement. Calibration of this equipment should be against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated or, when this is not possible, have an accuracy that assures the equipment being calibrated will be within required tolerance and that the basis of acceptance is documented and authorized by responsible management.

12.7 Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if it can be shown to be adequate for the requirements and the basis of acceptance is documented and authorized by responsible management.

12.8 Reference and transfer standards are traceable to nationally recognized standards; where national standards do not exist, provisions are established to document the basis for calibration.

12.9 Measures are taken and documented to determine the validity of previous inspections performed and the acceptability of items inspected or tested since the last calibration when measuring and test equipment is found to be out of calibration. Inspections or tests are repeated on items determined to be suspect.

Activities related to Handling, Storage, and Shipping (17.1.13) are acceptable if

13.1 Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by suitably trained individuals in accordance with predetermined work and inspection instructions.

13.2 Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity.

13.3 The description of the control of handling, storage, and shipping listed above satisfies the regulatory position in Regulatory Guide 1.38.

Activities related to Inspection, Test, and Operating Status (17.1.14) are acceptable if:

14.1 Procedures are established to indicate the inspection, test, and operating status of structures, systems, and components throughout fabrication, installation, and test.

14.2 Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps.

14.3 Procedures are established and described to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval.

14.4 The status of nonconforming, inoperative, or malfunctioning structures, systems, and components is documented and identified to prevent inadvertent use.

Activities related to Nonconforming Materials, Parts, or Components (17.1.15) are acceptable if:

15.1 Procedures are established and described for identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components and as applicable to services (including computer codes) if disposition is other than to scrap. The procedures provide for an independent review of nonconformances, including disposition and closeout.

15.2 Organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconforming items.

15.3 Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconformance, and the inspection requirements; and includes signature approval of the disposition. Nonconformances are corrected or resolved prior to the initiation of the preoperational test program on the item.

15.4 Reworked, repaired, and replacement items are inspected and tested in accordance with the original inspection and test requirements or acceptable alternatives.

15.5 Nonconformance reports are periodically analyzed to show quality trends, and the significant results are reported to management for review and assessment.

Activities related to Corrective Action (17.1.16) are acceptable if:

16.1 Procedures are established and described indicating an effective corrective action program has been established.

16.2 Corrective action is initiated following the determination of a condition adverse to quality (such as a nonconformance, failure, malfunction, deficiency, deviation, and defective material and equipment) to preclude recurrence.

16.3 Followup action is taken to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

16.4 Significant conditions adverse to quality, the cause of the conditions, and the corrective action taken to preclude repetition are documented and reported to cognizant levels of management for review and assessment.

Activities related to Quality Assurance Records (17.1.17) are acceptable if:

17.1 The scope of the records program is described. QA records include results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports; nonconformance reports; and corrective action reports.

17.2 Organizational responsibilities are described for the definition and implementation of activities related to QA records.

17.3 Inspection and test records contain the following where applicable:

a. A description of the type of observation.

b. The date and results of the inspection or test.

c. Information related to conditions adverse to quality.

d. Inspector or data recorder identification.

e. Evidence as to the acceptability of the results.

f. Action taken to resolve any discrepancies noted.

17.4 Suitable facilities for the storage of records are described and satisfy the regulatory position given in Regulatory Guide 1.88 (endorses N45.2.9). Alternatives to the fire protection rated provisions are acceptable if records storage facilities conform to NFPA No. 232 Class 1 for permanent-type records and that the 2-hour fire rating requirement contained in the proposed N45.2.9 standard is met by applicants in any one of the following three ways. Specifically, (1) a 2-hour vault meeting NFPA No. 232; (2) 2-hour rated file containers meeting NFPA No. 232 (Class B); or (3) a 2-hour rated fire resistant file room meeting NFPA No. 232 if the following additional provisions are provided.

1. Early warning fire detection and automatic fire suppression should be provided, with electronic supervision at a constantly attended central station.

2. Records should be stored in fully enclosed metal cabinets. Records should not be permitted on open steel shelving. No storage of records should be permitted on the floor of the facility. Adequate access and aisle ways should be maintained at all times throughout the facility.

3. Work not directly associated with records storage or retrieval should be prohibited within the records storage facility. Examples of such prohibited

activities include but are not limited to: records reproduction, film developing, and fabrication of microfiche cards.

4. Smoking and eating/drinking should be prohibited throughout the records storage facility.

5. Ventilation, temperature, and humidity control equipment should be protected inside with standard fire-door dampers where they penetrate fire barriers bounding the storage facility.

17.5 The description of the control of records provisions listed above satisfies the regulatory position of Regulatory Guide 1.88.

Activities related to Audits (17.1.18) are acceptable if:

18A1. Audits to assure that procedures and activities comply with the overall QA program are performed by:

a. The QA organization to provide a comprehensive independent verification and evaluation of quality-related procedures and activities.

b. The applicant (and principal contractors) to verify and evaluate the QA programs, procedures, and activities of suppliers.

18A2. An audit plan is prepared identifying audits to be performed, their frequencies, and schedules. Audits should be regularly scheduled based upon the status and safety importance of the activities being performed and are initiated early enough to assure effective QA during design, procurement, manufacturing, construction, installation, inspection, and testing.

18A3. Audits include an objective evaluation of quality-related practices, procedures, instructions; activities and items; and review of documents and records to ensure that the QA program is effective and properly implemented.

18A4. Provisions are established requiring that audits be performed in all areas where the requirements of Appendix B to 10 CFR Part 50 are applicable. Areas which are often neglected but should be included are activities associated with:

a. The determination of site features which affect plant safety (e.g., core sampling, site and foundation preparation, and methodology). (PSAR only).

b. The preparation, review, approval, and control of early procurements. (PSAR only).

c. Indoctrination and training programs.

d. Interface control among the applicant and the principal contractors.

e. Corrective action, calibration, and nonconformance control systems.

f. SAR and SSAR commitments.

18B1. Audit data are analyzed and the resulting reports indicating any quality problems and the effectiveness of the QA program, including the need for reaudit of deficient areas, are reported to management for review and assessment.

18B2. Audits are performed in accordance with pre-established written procedures or checklists and conducted by trained personnel having no direct responsibilities in the areas being audited.

18B3. The description of the conduct of audit provisions satisfies the requirements of ANSI Standard N45.2.12, Draft 3, Rev. 4 - February 1974.

III. REVIEW PROCEDURES

Each element of the QA program description will be reviewed against the acceptance criteria described in subsection II, including the Regulatory Guides, ANSI Standard and Branch Technical Position listed in subsection V. The reviewer's judgment during the review is to be based on an assessment of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved. Any exceptions or alternatives to this SRP section, including the regulatory positions presented in the Regulatory Guides and ANSI Standards in subsection V, will be carefully reviewed to assure that they are clearly defined and that an adequate basis exists for acceptance.

The acceptability of the QA program is determined by the following review procedures:

1. The QA program description is reviewed in detail to determine if each of the criteria of 10 CFR 50 Appendix B has been acceptably addressed and if there is an adequate commitment to comply with the regulatory positions in the appropriate issue of the Regulatory Guides and the requirements of the ANSI Standard in subsection V, as identified by number, title, revision or date. The QA program description is also reviewed to assure that the applicant's approach to meeting the QA criteria and commitments is acceptable.

2. The measures described to implement 10 CFR 50 Appendix B are evaluated for:

   a. Technical acceptability (i.e., do they meet the Regulations and Regulatory Guides?)

b. Workability (i.e., do they seem to fit into an overall plan of action that can be implemented?)

c. Management support (i.e., do QA program measures have adequate review, approval, and endorsement of management?)

This evaluation is based primarily on the acceptance criteria contained in subsection II.

3. The duties, responsibility, and authority of personnel performing QA functions are reviewed to assure they provide sufficient independence to effectively perform these functions.

4. Through review of information provided, meetings with the applicant, and by review of the Office of Inspection and Enforcement position statement and inspection reports, a judgment is made of the applicant's capability to carry out its QA responsibilities.

5. Satisfaction with program commitments and descriptions of how the commitments will be met, organizational arrangements, and capabilities to fulfill QA requirements should lead to the conclusion of acceptability, as described in subsection IV.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his review is sufficiently complete and adequate to support conclusions of the following type to be included in the staff's Safety Evaluation Report:

Based on our detailed review and evaluation of the QA program description contained in the (topical report or SAR) for (nuclear facility), we conclude that:

1. The organizations and persons performing QA functions have the required independence and authority to effectively carry out the QA program without undue influence from those directly responsible for costs and schedules.

2. The QA program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR Part 50; with the criteria contained in SRP Section 17.1; with the regulatory positions presented in the following Regulatory Guides; and with the requirements of the following ANSI Standard:

Reg. Guide/ANSI Std.	Title	Revision or Date

3. The QA program covers activities affecting structures, systems, and components important to safety as identified in the PSAR.

Accordingly, the staff concludes that the applicant's description of the QA program is in compliance with applicable NRC regulations and industry standards and can be implemented for the (specify) phases of (specify application).

V. REFERENCES

1. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. 10 CFR Part 50.55a, "Codes and Standards."

3. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

4. Regulatory Guide 1.8, "Personnel Selection and Training" (endorses ANSI/ANS 3.1).

5. Regulatory Guide 1.26, "Quality Group Classification, and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants."

6. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" (endorses N45.2).

7. Regulatory Guide 1.29, "Seismic Design Classification."

8. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (endorses N45.2.4).

9. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (endorses N45.2.1).

10. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (endorses N45.2.2).

11. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants" (endorses N45.2.3).

12. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (endorses N45.2.6).

13. Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (endorses N45.2.11).

14. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions" (endorses N45.2.10).

15. Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (endorses N45.2.9).

16. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (endorses N45.2.5).

17. Regulatory Guide 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (endorses N45.2.8).

18. Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (endorses N45.2.13).

19. ANSI N45.2.12 (Draft 3, Revision 4 - February 1974), "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," American National Standards Institute.

20. Branch Technical Position (BTP) ASB 9.5-1 (attached to SRP-Section 9.5.1).

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

SECTION 17.2 QUALITY ASSURANCE DURING THE OPERATIONS PHASE

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

1. AREAS OF REVIEW

QAB reviews and evaluates the applicant's operational quality assurance (QA) program as described in the FSAR. The review at the operating license stage addresses both the "offsite" and "onsite" QA controls to be applied to those activities that may affect the quality of items important to safety during the operation, maintenance, and modification of a nuclear power plant. The review covers the QA controls to be applied to those activities (e.g., designing, constructing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, maintaining, modifying, operating, inspecting, and testing) that may affect the quality of structures, systems, and components important to safety.

The review extends to the determination of how the applicable requirements of the 18 criteria of Appendix B to 10 CFR Part 50 are satisfied by the proposed QA program.

Where an NRC-accepted QA topical report is referenced in the application, the referenced QA program is not re-reviewed except for conformance to the applicable Regulatory Guides.

The review will not involve an evaluation of the QA program for the design and construction phase and, therefore, the QAP description for design and construction should not be addressed in the FSAR unless referenced as applicable for repair and modifications only during the operation phase.

The areas of review for this SRP section are the same as those described in SRP Section 17.1 except:

1. Organization (item 1) delete from part A: "including the applicant's organization and principal contractors (architect engineer, nuclear steam supply system vendor, constructor, and construction manager when other than the constructor)."

2. Audits (item 18) add a part C: "Provisions for the audit of operating activities important to safety independent of the operating organization."

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 applicant must establish a QA program for the operations phase, including activities such as operation, maintenance, and modification of the nuclear power plant, in accordance with Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The QA program description presented in the FSAR must discuss how each criterion of Appendix B will be met. The acceptance criteria used by the QAB to evaluate the program are listed below. The acceptance criteria include commitments to comply with the regulatory positions presented in the appropriate issue of the Regulatory Guides including the requirements of ANSI Standard N45.2.12 and the Branch Technical Position listed in subsection V of SRP Section 17.1. Thus, these commitments constitute an integral part of the QA program description and requirements. Exceptions and alternatives to these acceptance criteria may be taken by applicants provided adequate justification is given; and the QAB review allows for considerable flexibility in defining methods and controls for satisfying pertinent regulations. When the QA program description meets the acceptance criteria of this SRP section or provides acceptable exceptions or alternatives, the program is considered to be in compliance with pertinent NRC regulations. The review will ascertain that the commitments and the description of how the commitments are implemented, to the extent necessary, are objective and stated in inspectable terms.

The Organization (SRP Section 17.2.1) elements responsible for the QA program are acceptable if:

1. The criteria described in 17.1.1* are satisfied except for:

   a. Item 1A4.

   b. The organizational elements within the parenthesis in item 1A5 of 17.1.1 be expanded to include operations and maintenance.

   c. The requirements that principal contractors describe QA responsibilities be deleted in 17.1.1, Item 1A.6.

   d. The requirements that a QA position be identified for principal contractors as described in 17.1.1, Item 1B1, be deleted.

   e. "The person at the construction site responsible for directing and managing the site QA program..." described in 17.1.1, Item 1C3, be changed to "The person...responsible for...the onsite QA program," and continue on with remaining sentence starting with "has appropriate organizational...."

2. For the situation in which the "onsite" QA organization does not report to the "offsite" organization:

*Refers to the acceptance criteria given in subsection II of SRP Section 17.1.

a. A formal line of communication is established between the position responsible for the "onsite" QA program and the "offsite" QA program.

b. The "offsite" QA manager maintains an overview of "onsite" QA activities through audits, surveillance, and/or inspection, as further delineated in subsequent criteria.

The Quality Assurance Program (SRP Section 17.2.2) description is acceptable if:

1. The criteria described in 17.1.2 are satisfied except for:

   a. Item 2A1b.

   b. The requirement for the principal contractors to provide a commitment to comply with the regulatory positions in the Regulatory Guides and the endorsed ANSI standard addressed in Item 2B3.

   c. Item 2C2.

   d. Item 2C3.

2. Provisions are established for assuring the QA program for operations is implemented at least 90 days prior to fuel loading.

Activities related to Design Control (SRP Section 17.2.3) are acceptable if:

1. The criteria described in 17.1.3 are satisfied.

2. Measures are provided to assure that responsible plant personnel are made aware of design changes/modifications which may affect the performance of their duties.

Activities related to Procurement Document Control (17.2.4) are acceptable if:

1. The criteria described in 17.1.4 are satisfied.

Activities related to Instructions, Procedures, and Drawings (17.2.5) are acceptable if:

1. The criteria described in 17.1.5 are satisfied.

Activities related to Document Control (17.2.6) are acceptable if:

1. The criteria described in 17.1.6 are satisfied.

2. In the case where the "onsite" QA organization does not report to the "offsite" QA organization, the "offsite" QA organization should review and concur with the "onsite" QA program.

3. Maintenance, modification and inspection procedures are reviewed by qualified personnel knowledgeable in QA disciplines (normally the QA organization) to determine:

   a. The need for inspection, identification of inspection personnel, and documentation of inspection results.

   b. That the necessary inspection requirements, methods, and acceptance criteria have been identified.

Activities related to Control of Purchased Material, Equipment, and Services (17.2.7) are acceptable if:

1. The criteria described in 17.1.7 are satisfied.

Activities related to Identification and Control of Materials, Parts, and Components (17.2.8) are acceptable if:

1. The criteria described in 17.1.8 are satisfied.

Activities related to the Control of Special Processes (17.2.9) are acceptable if:

1. The criteria described in 17.1.9 are satisfied.

Activities related to Inspection (17.2.10) are acceptable if:

1. The criteria described in 17.1.10 are satisfied.

2. When inspections associated with normal operations of the plant (such as routine maintenance, surveillance, and tests) are performed by individuals other than those who performed or directly supervised the work, but are within the same group, the following controls are met:

   a. The quality of the work can be demonstrated through a functional test when the activity involves breaching a pressure retaining item.

   b. The qualification criteria for inspection personnel are reviewed and found acceptable by the QA organization prior to initiating the inspection.

Activities related to Test Control (17.2.11) are acceptable if:

1. The criteria described in 17.1.11 are satisfied.

Activities related to Control of Measuring and Test Equipment (17.2.12) are acceptable if:

1. The criteria described in 17.1.12 are satisfied.

Activities related to Handling, Storage, and Shipping (17.2.13) are acceptable if:

1. The criteria described in 17.1.13 are satisfied.

2. Provisions are described for the storage of chemicals, reagents (including control of shelf life), lubricants, and other consumable materials.

Activities related to Inspection, Test, and Operating Status (17.2.14) are acceptable if:

1. The criteria described in 17.1.14 are satisfied.

Activities related to Nonconforming Materials, Parts, or Components (17.2.15) are acceptable if:

1. The criteria described in 17.1.15 are satisfied.

Activities related to Corrective Action (17.2.16) are acceptable if:

1. The criteria described in 17.1.16 are satisfied.

Activities related to Quality Assurance Records (17.2.17) are acceptable if:

1. The criteria described in 17.1.17 are satisfied.

2. QA records include operating logs, maintenance and modification procedures and related inspection results, reportable occurrences, and other records required by Technical Specifications.

Activities related to Audits (17.2.18) are acceptable if:

1. The criteria described in 17.1.18 are satisfied.

2. Where the "onsite" QA organization does not report to the "offsite" organization:

   a. The "offsite" QA organization conducts audits sufficient to verify adequacy of activities conducted by the "onsite" QA organization.

   b. The "offsite" QA organization reviews and concurs in the schedule and scope of audits performed by the "onsite" QA organization.

   c. Results of audits performed by the "onsite" QA organization are provided to the "offsite" QA organization for review and assessment.

III. REVIEW PROCEDURES

Same as SRP Section 17.1 except that the Office of Inspection & Enforcement (I&E) does not provide a position statement to QAB relative to their assessment of the QA program implementation for SER input. I&E provides this assessment to the Licensing Project Manager.

IV. EVALUATION FINDINGS

Same as SRP Section 17.1.

V. REFERENCES

Same as SRP Section 17.1 except replace item 6, Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" (endorses N45.2) with Regulatory Guide 1.33 "Quality Assurance Program Requirements (Operation)" (endorses N18.7).