12. RADIATION PROTECTION

This chapter of the SAR should provide information on methods for radiation protection and on estimated occupational radiation exposures to operating and construction personnel during normal operation and anticipated operational occurrences (including refueling; purging; fuel handling and storage; radioactive material handling, processing, use, storage, and disposal; maintenance; routine operational surveillance; inservice inspection; and calibration). It should provide information on facility and equipment design, the planning and procedures programs, and the techniques and practices employed by the applicant in meeting the standards for protection against radiation of 10 CFR Part 20 and the guidance given in the appropriate regulatory guides, where the practices set forth in such guides will be used to implement NRC regulations. Reference to other chapters for information needed in this chapter should be specifically made where required.

12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

12.1.1 Policy Considerations

Describe the management policy and organizational structure related to ensuring that occupational radiation exposures are ALARA. Describe the applicable responsibilities and the related activities to be conducted by the management individuals having responsibility for radiation protection and the policy of maintaining occupational exposures ALARA. In the PSAR, describe policy with respect to designing and constructing the plant; in the FSAR, describe the ALARA policy as it will be applied to plant operations. In the PSAR, indicate whether, and if so how, the ALARA policy guidance given in Section C.1 of Regulatory Guide 8.8 (Ref. 1) and in Regulatory Guides 8.10 (Ref. 2) and 1.8 (Ref. 3) will be followed; if it will not be followed, describe the specific alternative approaches to be used. Indicate how the requirements of 10 CFR Part 20 (Ref. 4) will be met.

12.1.2 Design Considerations

In the PSAR, describe how experience from past designs and from operating plants is used to develop improved design for ensuring that occupational radiation exposures are ALARA. Describe how ALARA design guidance (both general and specific) is given to the individual designers. Describe how the design is directed toward reducing the need for maintenance of equipment and to reducing radiation levels and time spent where maintenance and other operational activities are required. Describe any mechanisms that provide for design review by a competent professional in radiation protection such as the utility radiation protection manager. These descriptions should be detailed in the PSAR, including an indication of whether, and if so how, the design consideration guidance provided in Section C.1 of Regulatory Guide 8.8 will be followed; if it will not be followed, describe the specific alternative approaches to be used.

The detailed facility design features for radiation protection and for ensuring that occupational radiation exposures will be ALARA should be covered in Section 12.3.1.

12.1.3 Operational Considerations

In the PSAR, describe the methods to be used to develop the detailed operational plans and procedures for ensuring that occupational radiation exposures are ALARA. Describe how these operational plans and procedures will impact on the design of the facility and how such planning has incorporated information from operating plant experience, other designs, etc. Describe how operational requirements are reflected in the design considerations described in Section 12.1.2 and the radiation protection design features described in Section 12.3.1. Indicate the extent to which the guidance on operational considerations given in Section C.1 of Regulatory Guide 8.8 and in Regulatory Guide 8.10 will be followed; if the guidance will not be followed, describe the specific alternative approaches to be used.

In the FSAR, provide the criteria and/or conditions under which various operating procedures and techniques for ensuring that occupational radiation exposures are ALARA are implemented for all systems that contain, collect, store, or transport radioactive liquids, gases, and solids (including, for example, the turbine system (for BWRs); the nuclear steam supply system; the residual heat removal systems; the spent fuel transfer, storage, and cleanup systems; and the radioactive waste treatment, handling, and storage systems). Describe means for planning and developing procedures for such radiation-exposure-related operations as maintenance, inservice inspections, radwaste handling, and refueling in a manner that will ensure that the exposures are ALARA. Describe any changes in operating procedures that result from the ALARA operational procedures review.

12.2 Radiation Sources

12.2.1 Contained Sources

In the PSAR, the sources of radiation that are the bases for the radiation protection design should be described in the manner needed as input to the shield design calculation. Those sources that are contained in equipment of the radioactive waste management systems should be described. In this section, source descriptions should be provided for other sources such as the reactor core, the spent fuel storage pool, various auxiliary systems, the steam lines and turbine system (including reheaters, moisture separators, etc.) as sources of N-16 in a BWR, and the equipment, systems, and piping containing activation product sources. For the reactor core, describe the source as it is used to determine radiation levels external to the biological shield at locations where occupancy may be required. For other sources, the description should tabulate sources by isotopic composition or gamma ray energy groups, strength (curie content), and geometry, as well as provide the basis for the values. The source location in the plant should be specified so that all important sources of radioactivity can be located on plant layout drawings. For all the sources identified above, including activation product sources, the models and parameters for calculating the source magnitudes should be provided. Indicate whether, and if so how, the applicable guidance provided in ANSI N237 (Ref. 5) has been followed; if not followed, describe the specific alternative methods used. Describe any required byproduct, source, and special nuclear material (Refs. 6, 7, and 8) that may require shielding design considerations. In the FSAR, provide a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 100 millicuries. Provide additional details (and any changes) of source descriptions that are used to develop the final shield design.

12.2.2 Airborne Radioactive Material Sources

In the PSAR, the sources of airborne radioactive material in equipment cubicles, corridors, and operating areas normally occupied by operating personnel should be described in the manner required for design of personnel protective measures and dose assessment. Those airborne radioactivity sources that have to be considered for their contribution to the plant effluent releases through the radioactive waste management system or the plant ventilation systems should be described in Chapter 11. Any other sources of airborne radioactivity in the areas mentioned above that are not covered in Chapter 11 should be included and described here. Sources resulting from reactor vessel head removal, relief valve venting, and movement of spent fuel should be included. The description should include a tabulation of the calculated concentrations of airborne radioactive material by nuclides expected during normal operation and anticipated operational occurrences for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. The models and parameters for calculating airborne radioactivity concentrations should be provided. In the FSAR, describe any changes or additions to the source data since the PSAR.

12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

In the PSAR, describe equipment and facility design features used for ensuring that occupational radiation exposures are ALARA. Indicate whether, and if so how, the design feature guidance given in Section C.2 of Regulatory Guide 8.8 has been followed; if not followed, describe the specific alternative approaches used.

Provide illustrative examples of the facility design features used in the PSAR design stage as applied to the systems listed in Section 12.1.3. The description should include those features that reduce 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 any other features that reduce radiation exposure of personnel. It should include descriptions of methods for reducing the production, distribution, and retention of activation products through design methods, material selection, water chemistry, decontamination procedures, etc. An illustrative example should be provided for each of the following components (including equipment and piping layouts): liquid filters, demineralizers, absorber beds, particulate filters, recombiners, tanks, evaporators, pumps, steam generators, valve operating stations, and sampling stations. In the FSAR, the location of sampling ports, instrumentation, and control panels should be provided.

In the PSAR, provide scaled layout and arrangement drawings of the facility showing the locations of all sources described in Section 12.2. Provide on the layouts the radiation zone designations, including zone boundaries for both normal operational and refueling outage conditions. Reference other chapters as appropriate. The layouts should show shield wall thicknesses, controlled access areas, personnel and equipment decontamination areas, contamination control areas, traffic patterns, location of the health physics facilities, location* of airborne radioactivity and area radiation monitors, location of control panels for radwaste equipment and components, location of the onsite laboratory for analysis of chemical and radioactivity samples, and location of the counting room. Specify the design basis radiation level in the counting room during normal operation and anticipated operational occurrences. Describe the facilities and equipment such as hoods, glove boxes, filters, special handling equipment, and special shields that are related to the use of sealed and unsealed special nuclear, source, and byproduct material. In the FSAR, describe changes or additions to the radiation protection design since the PSAR.

12.3.2 Shielding

In the PSAR, provide information on the shielding for each of the radiation sources identified in Chapter 11 and Section 12.2, including the criteria for penetrations, the material, the method by which the shield parameters (cross sections, buildup factors, etc.) were determined, and the assumptions, codes, and techniques used in the calculations. Describe special protective features that use shielding, geometric arrangement (including equipment separation), or remote handling to ensure that occupational radiation exposures will be ALARA in normally occupied areas such as valve operating stations and sample collection stations. Indicate whether, and if so how, the guidance provided in Regulatory Guide 1.69 (Ref. 9) on concrete radiation shields and in Regulatory Guide 8.8 on special protective features has been followed; if not followed, describe the specific alternative methods used. In the FSAR, describe changes or additions in the shielding since the PSAR.

12.3.3 Ventilation

In the PSAR, the personnel protection features incorporated in the design of the ventilation system should be described. Those aspects of the design that relate to removing airborne radioactivity from equipment

*In the PSAR, if available, and update in the FSAR.

cubicles, corridors, and operating areas normally occupied by operating personnel and into the effluent control systems should be described in Chapter 11. Describe here any ventilation system protective features not covered in Chapter 11 or provided by the descriptions in Chapter 9. Include those aspects of the systems that relate to controlling the concentration of radioactivity in the areas mentioned above. Provide an illustrative example of the air cleaning system design, including an example layout of an air cleaning system housing showing filter mountings, access doors, aisle space, service galleries, and provisions for testing, isolation, and decontamination. Provide the criteria established for the change-out of air filters and adsorbers in the air cleaning system. Indicate whether, and if so how, the applicable guidance provided in Regulatory Guide 1.52 (Ref. 10) has been followed; if not followed, describe the specific alternative methods used. In the FSAR, include any changes or additions in the ventilation system design protective features since the PSAR.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

In the PSAR, describe the fixed area radiation and continuous airborne radioactivity monitoring instrumentation and the criteria for selection and placement.

In the FSAR, provide information on the auxiliary and/or emergency power supply and the range, sensitivity, accuracy, precision, calibration methods and frequency, alarm setpoints, recording devices, and location of detectors, readouts, and alarms for the monitoring instrumentation. Accident considerations and other needs for high range instrumentation should be included. In the FSAR, provide the location of airborne monitor sample collectors, and give details of sampling lines and pump location.

In the PSAR, describe the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations, including airborne radioiodines and other radioactive materials, from the area being sampled.

In the FSAR, describe the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of §70.24 of 10 CFR Part 70 for the storage area for new fuel.

Indicate whether, and if so how, the guidance provided by Regulatory Guides 1.21 (Ref. 11), 8.2 (Ref. 12), 8.8, 8.12 (Ref. 13), and 1.97 (Ref. 14) and ANSI N13.1-1969 (Ref. 15) has been followed; if not followed, describe the specific alternative methods used.

12.4 Dose Assessment

In the PSAR, provide the estimated annual occupancy (including numbers of personnel and durations of occupancy) of the plant radiation areas during normal operation and anticipated operational occurrences, including, for example, maintenance, repairs, replacement of pump and valves, and plugging of steam generator tubes. For areas with expected airborne radioactivity concentrations (discussed in Section 12.2.2) during normal operation and anticipated operational occurrences as discussed above, provide estimated man-hours of occupancy and estimated inhalation exposures to personnel. Provide the objectives and criteria for design dose rates in various areas and an estimate of the annual man-rem doses associated with major functions such as operation, normal maintenance, radwaste handling, refueling, and inservice inspection. The basis, models, and assumptions for the above values should be provided. For routine or repetitive activities expected to occur with reasonably predictable frequencies and involving well-known sequences of operations, dose assessment should include, to the extent practicable, consideration of the specific plant and operation and should consider actual estimated dose rates at the various locations.

In the FSAR, provide updated estimates of annual man-rem doses for the functions listed above and the assumptions used in determining these values. Describe any changes made during planning or design review for the purpose of reducing these projected doses. Actual exposure data from similar operating plants operated in a similar manner may be used for the dose assessment for unpredictable activities but should be corrected for improvements in plant design and operating procedures.

In the PSAR, provide the estimated annual dose at the boundary of the restricted area (as defined in 10 CFR §20.3), at the site boundary, and, for multi-unit plants, at various locations in a new unit construction area from onsite radiation sources such as the turbine systems (for BWRs), the auxiliary building, the reactor building, and stored radioactive wastes and from radioactive effluents (direct radiation from the gaseous radioactive effluent plume). Provide estimated annual doses to construction workers due to radiation from these sources from the existing operating plant(s), and the annual man-rem doses associated with such construction. Include models, assumptions, and input data. In the FSAR, changes or additions since the PSAR should be provided. Indicate whether, and if so how, the guidance provided by Regulatory Guide 8.19 (Ref. 16) is followed; if not followed, describe the specific alternative methods used.

12.5 Health Physics Program

12.5.1 Organization

In the PSAR, describe the administrative organization of the health physics program, including the authority and responsibility of each position identified. Indicate whether, and if so how, the guidance of Regulatory Guides 8.2, 8.8, 8.10, and 1.8 has been followed; if not followed, describe the specific alternative approaches used. In the FSAR, describe the experience and qualification of the personnel responsible for the health physics program and for handling and monitoring radioactive materials, including special nuclear, source, and byproduct materials. Reference Chapter 13 as appropriate.

12.5.2 Equipment, Instrumentation, and Facilities

In the PSAR, provide the criteria for selection of portable and laboratory technical equipment and instrumentation for performing radiation and contamination surveys, for airborne radioactivity monitoring and sampling, for area radiation monitoring, and for personnel monitoring during normal operation, anticipated operational occurrences, and accident conditions. Describe the instrument storage, calibration, and maintenance facilities. Describe and identify the location of the health physics facilities (including locker rooms, shower rooms, offices, and access control stations), laboratory facilities for radioactivity analyses, protective clothing, respiratory protective equipment, decontamination facilities (for equipment and personnel), and other contamination control equipment and areas that will be available. Indicate whether, and if so how, the guidance provided by Regulatory Guides 8.3 (Ref. 17), 8.4 (Ref. 18), 8.8, 8.9 (Ref. 19), 8.12, 8.14 (Ref. 20), 8.15 (Ref. 21), and 1.97 has been followed; if not followed, describe the specific alternative methods used. In the FSAR, provide the location of the respiratory protective equipment, protective clothing, and portable and laboratory technical equipment and instrumentation. Describe the type of detectors and monitors and the quantity, sensitivity, range, and frequency and methods of calibration for all the technical equipment and instrumentation mentioned above.

12.5.3 Procedures

In the FSAR, the policy, methods, frequencies, and procedures for conducting radiation surveys should be described. Describe the procedures and methods of operation that have been developed for ensuring that occupational radiation exposures will be ALARA. Include a description of the procedures used in refueling, inservice inspections, radwaste handling, spent fuel handling, loading and shipping, normal operation, routine maintenance, and sampling and calibration that are specifically related to ensuring the radiation exposures will be ALARA. Describe the physical and administrative measures for controlling access and stay time for radiation areas. Reference may be made to Section 12.1, as appropriate. Describe the bases and methods for monitoring and control of contamination of personnel, equipment, and surface. Radiation protection training programs should be described. Indicate whether, and if so how, the guidance given in Regulatory Guides 8.2, 8.7 (Ref. 22), 8.8, 8.9, 8.10, 8.13 (Ref. 23), 1.8, 1.16 (Ref. 24), 1.33 (Ref. 25), and 1.39 (Ref. 26) will be followed; if it will not be followed, describe the specific alternative approaches to be used. Reference Chapter 13 as appropriate. Indicate how the requirements of 10 CFR Part 19 (Ref. 27) will be met.

Describe the methods and procedures for personnel monitoring (external and internal), including methods of recording, reporting, and analyzing results. Describe the program for internal radiation exposure assessment (whole body counting and bioassay), including the bases for selecting personnel who will be in the program, the frequency of their whole-body count and bioassay, and any nonroutine bioassay that will be performed.

Describe the methods and procedures for evaluating and controlling potential airborne radioactivity concentrations. Discuss any requirements for special air sampling and the issuance, selection, use, and maintenance of respiratory protective devices, including training programs and respiratory protective equipment fitting programs.

Method of handling and storage of sealed and unsealed byproduct, source, and special nuclear material should be described.

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. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable."
3. Regulatory Guide 1.8, "Personnel Selection and Training."
4. 10 CFR Part 20, "Standards for Protection Against Radiation."
5. ANSI N237, "Source Term Specification," Final Draft, 1977.
6. 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material."
7. 10 CFR Part 40, "Domestic Licensing of Source Material."
8. 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."
9. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."
10. 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."
11. 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."
12. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."
13. Regulatory Guide 8.12, "Criticality Accident Alarm Systems."
14. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

15. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

16. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants--Design Stage Man-Rem Estimates."

17. Regulatory Guide 8.3, "Film Badge Performance Criteria."

18. Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters."

19. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

20. Regulatory Guide 8.14, "Personnel Neutron Dosimeters."

21. Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

22. Regulatory Guide 8.7, "Occupational Radiation Exposure Records Systems."

23. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

24. Regulatory Guide 1.16, "Reporting of Operating Information--Appendix A Technical Specifications."

25. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

26. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

27. 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections."

13. CONDUCT OF OPERATIONS

This chapter of the SAR should provide information relating to the preparations and plans for operation of the plant. Its purpose is to provide assurance that the applicant will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed by the licensee are adequate to protect public health and safety.

The information required at the PSAR stage pursuant to 10 CFR §50.34(a)(6), (9), and (10) should demonstrate adequate planning for the operational phase of the plant. The information required at the FSAR stage pursuant to 10 CFR §50.34(b)(6i), (6iv), (6v), and (7) should provide firm evidence that operating phase plans have been or are being implemented.

13.1 Organizational Structure Of Applicant

13.1.1 Management and Technical Support Organization

The description in this section of the corporate or home-office organization, its functions and responsibilities, and the number and the qualifications of personnel should be directed to activities that include facility design, design review, design approval, construction management, testing, and operation of the plant. The following specific information should be included.

13.1.1.1 Design and Operating Responsibilities. In the PSAR, the description should include the corporate functions and their specific responsibilities for the activities described in items 1 and 2 below and plans relative to item 3 below. In the FSAR, the description should summarize the degree to which the activities described in items 1 and 2 below have been accomplished, provide a schedule for completing these activities, and describe the specific responsibilities and activities relative to item 3 below.

1. Design and Construction Activities (Project Phase). The extent and assignment of these activities are generally contractual in nature and determined by the applicant. (Quality assurance aspects should be described in Section 17.1.) The following 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,

c. Review and approval of plant design features,

d. Site layout with respect to environmental effects and security provisions,

e. Development of safety analysis reports,

f. Review and approval of material and component specifications,

g. Procurement of materials and equipment, and

h. Management and review of construction activities.

2. Preoperational Activities. These are the activities that should be substantially accomplished before preoperational testing begins and generally before submittal of the FSAR. The following should be included:

a. Development of human engineering design objectives and design phase review of proposed control room layouts,

b. Development and implementation of staff recruiting and training programs,

c. Development of plans for initial testing, and

d. Development of plant maintenance programs.

3. Technical Support for Operations. Technical services and backup support for the operating organization should become available prior to the initial testing program and continue throughout the life of the plant. The following are special capabilities that should be included:

a. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, and instrumentation and controls engineering,

b. Plant chemistry,

c. Health physics,

d. Fueling and refueling operations support, and

e. Maintenance support.

13.1.1.2 Organizational Arrangement. In the PSAR, the description should include organization charts reflecting the current headquarters and engineering structure and any planned modifications and additions to reflect the added functional responsibilities (described in 13.1.1.1) associated with the addition of the nuclear plant to the applicant's power generation capacity. The description should show how these responsibilities are delegated and assigned within and from the headquarters staff and the number of persons assigned or expected to be assigned to each of the working or performance level organizational units identified to implement these responsibilities.

In the FSAR, the description should include organization charts reflecting the current corporate structure and the specific working or performance level organizational units that will provide technical support for operation (Section 13.1.1.1, item 3). If these functions are to be provided from outside the corporate structure, the contractual arrangements should be described.

13.1.1.3 Qualifications. The PSAR should describe general qualification requirements in terms of educational background and experience requirements for positions or classes of positions identified in 13.1.1.2. Personnel resumes should be provided for assigned persons identified in 13.1.1.2 holding key or supervisory positions in disciplines or job functions unique to the nuclear field or this project. 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.

The FSAR should identify qualification requirements for headquarters staff personnel, which should be described in terms of educational background and experience requirements, for each identified position or class of positions providing headquarters technical support for operations. In addition, the FSAR should include resumes of individuals already employed by the applicant to fulfill responsibilities identified in item 3 of Section 13.1.1.1, including that individual whose job position corresponds most closely to that identified as "engineer in charge."

13.1.2 Operating Organization

This section of the SAR should describe the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant. The following specific information should be included.

13.1.2.1 Plant Organization. Provide an organization chart showing the title of each position, the number of persons assigned to common or duplicate positions (e.g., technicians, shift operators, repairmen), 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 planned changes and additions as new units are added to the station. The schedule, relative to the fuel loading date for each unit, for filling all positions should be provided.

13.1.2.2 Plant Personnel Responsibilities and Authorities. 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. Unlicensed operators,
6. Technical supervision,
7. Nuclear engineering supervision,
8. Radiation protection supervision,
9. Instrumentation and controls engineering supervision,
10. Instrumentation and controls maintenance supervision,
11. Equipment maintenance supervision, and
12. Quality assurance and quality control supervision.

For each position, where applicable, required interfaces with offsite personnel or positions identified in 13.1.1 should be described. Such interfaces include defined lines of reporting responsibilities, e.g., from the plant manager to his immediate supervisor, as well as functional or communication channels. In the FSAR, the following should also be described:

1. The line of succession of authority and responsibility for overall station operation through at least three persons, in the event of unexpected contingencies of a temporary nature, and

2. The delegation of authority 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, 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 and the proportion of their time that they will routinely and nonroutinely be assigned to the other unit.

13.1.2.3 Operating Shift Crews. The position titles, applicable operator licensing requirements for each, and the minimum numbers of personnel planned for each shift should be described for all combinations of units proposed to be at the station in either operating or cold shutdown mode. Also describe shift crew staffing plans unique to refueling operations. In addition, the proposed means of assigning shift responsibility for implementing the radiation protection program on a round-the-clock basis should be described.

13.1.3 Qualifications of Nuclear Plant Personnel

13.1.3.1 Qualification Requirements. This section of the SAR should describe the education, training, and experience requirements established for each management, operating, technical, and maintenance position category in the operating organization described in Section 13.1.2. Regulatory Guide 1.8, "Personnel Selection and Training," contains guidance on selection and training of personnel. The SAR should specifically indicate a commitment to meet the regulatory position stated in this guide or provide an acceptable alternative. Where a clear correlation cannot be made between the proposed plant staff positions and those referenced by Regulatory Guide 1.8, each position on the plant staff should be listed along with the corresponding position referenced by Regulatory Guide 1.8, or with a detailed description of the proposed qualifications for that position.

13.1.3.2 Qualifications of Plant Personnel (FSAR). The qualifications of the initial appointees to (or incumbents of) plant positions should be presented in resume format for key plant managerial and supervisory personnel through the shift supervisory level. The resumes should identify individuals by position title and, as a minimum, describe the individual's formal education, training, and experience (including any prior AEC or NRC licensing).

13.2 Training

13.2.1 Plant Staff Training Program

The PSAR should provide a description of the proposed training program in nuclear technology and other subjects important to safety for the entire plant staff. The FSAR should describe the training program as actually carried out up to the time of FSAR preparation and should note any significant changes from the program described in the PSAR. Regulatory Guide 1.8, "Personnel Selection and Training," provides guidance on an acceptable basis for relating initial training programs to plant staff positions. The PSAR and FSAR should indicate whether this guidance will be followed. If such guidance will not be followed, specific alternative methods that will be used should be described along with a justification for their use. A list of Commission regulations, guides, and reports pertaining to training of licensed and unlicensed nuclear power plant personnel is provided in Section 13.2.3.

13.2.1.1 Program Description. The program description should include the following information with respect to the formal training program in nuclear technology and other subjects important to safety (related technical training) for all plant management and supervisory personnel, Licensed Senior Operator (SRO) and Licensed Operator (RO) candidates, technicians, and general employees.

The PSAR should include:

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 which the course is given.

2. A description of proposed reactor operations experience training by nuclear power plant simulator or by assignment to a similar plant, including length of time (weeks), identity of simulator or plant, and identification by position of personnel to be trained.

3. A commitment to conduct an onsite formal training program and on-the-job training before initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to §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 commitment to conduct an initial fire protection training program for the plant staff including:

a. Provisions for drills during construction, and

b. Provisions for indoctrination of construction personnel, as necessary.

The initial training should be completed prior to receipt of fuel at the site.

6. A detailed description of the training program for the individual(s) responsible for formulating and ensuring the implementation of the fire protection program. The training program should be consistent with the information on fire protection systems provided in Section 9.5.1.

7. Means for evaluating the training program effectiveness for all employees. For individuals seeking an operator license prior to criticality, this includes the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to paragraph 55.25(b) of 10 CFR Part 55.

The FSAR should include:

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. A description of reactor operations experience training by nuclear power plant simulator or by assignment to a similar plant, including length of time (weeks), identity of simulator or plant, and identification by position of personnel to be trained.

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 title 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 §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 should describe:

a. The training planned for each member of the fire brigade,

b. The 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, and

e. The training provided for the fire protection staff members.

The description should include the course of instruction, the number of hours of each course, and the organization conducting the training.

6. Means for evaluating the training program effectiveness for each employee. For individuals seeking an operator license prior to criticality, this includes the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to paragraph 55.25(b) of 10 CFR Part 55.

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading. The PSAR should include a chart that shows the schedule of each part of the training program for each functional group of employees in the organization in relation to the schedule for preoperational testing, expected fuel loading, expected time for examinations prior to plant criticality for licensed operators, and expected time for examinations for licensed operators following plant criticality. In the FSAR, the applicant should include in the chart contingency plans for individuals applying for licenses prior to criticality in the event fuel loading is substantially delayed from the date indicated in the FSAR.

In the FSAR, the chart should reflect the extent to which the training program has been accomplished as of the approximate time of submittal of the FSAR.

13.2.2 Replacement and Retraining (FSAR)

This section should describe the applicant's plans for retraining of the plant staff, including requalification training for licensed operators and a commitment to provide training for replacement personnel.

13.2.2.1 Licensed Operators - Requalification Training. A detailed description of the applicant's licensed operator requalification training program should be provided. This description should show how the program will implement the requirements of Appendix A, "Requalification Programs for Licensed Operators of Production and Utilization Facilities," to 10 CFR Part 55.

13.2.2.2 Refresher Training for Unlicensed Personnel. The additional position categories on the plant staff for which retraining will be provided should be identified, and the nature, scope, and frequency of such retraining should be described.

13.2.2.3 Replacement Training. The applicant should briefly describe the training program for replacement personnel.

13.2.3 Applicable NRC Documents

The NRC regulations, regulatory guides, and reports listed below provide information pertaining to the training of nuclear power plant personnel. The SAR should indicate the extent to which the applicable portions of the guidance provided will be used and should justify any exceptions. Material discussed elsewhere in the SAR may be referenced.

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 and Training."

5. Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants."

6. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."

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

8. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable."

9. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

10. "Utility Staffing and Training for Nuclear Power," WASH-1130, Revised June 1973.

11. NRC Operator Licensing Guide, NUREG-0094, July 1976.

13.3 Emergency Planning

This section of the SAR should describe the applicant's plans for coping with emergencies pursuant to paragraphs (a)(10) and (b)(6)(v) of §50.34 of 10 CFR Part 50. The items to be discussed are set forth in Appendix E, "Emergency Plans for Production and Utilization Facilities," to 10 CFR Part 50. Guidance is provided in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants." The information provided should also contribute to a determination that the exclusion area and the low population zone for the site comply with the definitions of paragraphs (a) and (b) of §100.3 of 10 CFR Part 100.

13.3.1 Preliminary Planning (PSAR)

At the PSAR stage, the items requiring description are set forth in the introductory paragraph and paragraphs A through G of Section II of Appendix E to 10 CFR Part 50. The following statements clarify information requirements applicable to the paragraphs indicated.

With respect to paragraph B, the PSAR should identify the agency with primary responsibility for emergency preparedness planning for situations involving real or potential radiological hazards in the State where the facility is to be located. The arrangements that the applicant has made with this agency for coordinating emergency response plans for the environs of the plant should be explained. Similar arrangements with the appropriate agency of any neighboring State should be described if any part of the neighboring State is within the low population zone (LPZ). Arrangements made or to be made to accommodate the radiological emergency preparedness responsibilities of neighboring States whose borders are beyond the LPZ but still within approximately 5 miles of the plant should also be described. Other State agencies that are expected to have emergency response roles should be identified. Local government organizations (agencies) having jurisdictional authority and responsibility for emergency response within the boundaries of the proposed LPZ should be identified. The PSAR should also identify the nearest Regional Coordinating Office for the U.S. Department of Energy's Radiological Assistance Program, and any Federal agency and appropriate regional or local office thereof, having a jurisdictional responsibility on lands or waterways within the boundaries of the proposed LPZ.

With respect to paragraph C, measures proposed to be taken to cope with the following emergency situations should be succinctly described:

a. Personnel injuries occurring at the plant site,

b. Fire emergencies,

c. Severe natural phenomena in the environment, and

d. Accidents that could lead to small, moderate, or substantial releases of radioactivity to the environment.

With respect to such releases, the discussion should, in particular, address accident assessment measures to be taken by personnel within the plant, with emphasis on the methodology to be employed for real-time estimates of the potential consequences (e.g., projected doses) and criteria for notifying appropriate offsite authorities who may need to implement protective actions. References may be made to other parts of the PSAR for relevant information.

The PSAR should also confirm that one of the protective measures that is to be incorporated in coordinated emergency plans is evacuation of persons from the exclusion area and from any other potentially affected sector of the environs extending at a minimum to the outer boundary of the proposed LPZ. An analysis, including specific information and findings that will be needed to ensure the development of adequately coordinated emergency plans with respect to evacuation as a protective measure, should be provided, including the following:

1. Plots showing projected ground-level doses for stationary individuals, for both whole body and thyroid, resulting from the most serious design basis accident analyzed in the SAR. These should be based on the same isotopic release rates to the atmosphere (most conservative case) as those used in Chapter 15 of the PSAR for the purpose of showing conformance to the guideline dose criteria of 10 CFR Part 100. Relative concentrations (χ over Q) for the first 2 hours should be the same as those used for establishing conformance to the siting criteria of 10 CFR Part 100. Dose contributions for subsequent time intervals and for distances beyond the exclusion area boundary should reflect reasonable time averaging of calculated relative concentrations, plume front transit times, and radioactive decay in transit. Dose conversion factors should be based on Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I;" if not, the bases selected should be described. The bases for developing the plots should be fully described. These data should be presented in the following format:

a. Use an appropriate scale with time (hours) following onset of release as the ordinate and distance (miles) from the release point as the abscissa. Sufficient background grid lines should be included to permit reasonable interpolation by eye.

b. Provide curves for whole body doses of 1, 5, and 25 rem and thyroid doses of 5, 25, 150, and 300 rem. Each curve should represent an estimate of the elapsed time to reach the specified dose level as a function of distance from the release point under the conditions postulated.

c. Extend each curve to an ordinate of not less than 8.0 hours either from an ordinate of 2.0 hours or from an abscissa equal to the exclusion radius, whichever results in the greater range of coverage. If any such curve does not intersect the outer LPZ boundary, it should be extended to such intersection or to an elapsed time of 24 hours, whichever occurs first.

2. The expected accident assessment time. This figure should incorporate the time required to identify and characterize this accident, the time needed to predict the projected doses resulting from the accident, and the time to notify offsite authorities. Include sufficient information to support the estimate. Reference should be made to other parts of the PSAR, as necessary, where the subject matter of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," is addressed.

3. An estimate of the minimum elapsed time that offsite authorities might require before an initial warning to the public can be given.

4. An estimate of the elapsed time from the first warning to a member of the public that may be required to warn all resident and transient persons within the potential evacuation areas determined in item 5

below. Discuss the means that might be employed to provide such warnings for both daytime and nighttime conditions.

5. An estimate of elapsed times, measured from the time of initial warning to persons, to evacuate (a) the exclusion area and (b) defined sectors of the environs. Sectors of the environs chosen for this analysis may be bounded by geographical or man-made features but should generally cover an arc of not less than 45° centered on the plant. They should extend outward at least to the outer boundary of the proposed LPZ.*

6. Information that should be provided in support of the estimates of item 5 above include:

a. A map showing all roads available for vehicular evacuation of the exclusion area and environs extending at least 10 miles from the plant. Road network information to be shown on or keyed to the map should include the character of each road, all intersections, the number of lanes, whether improved or unimproved, and other factors that may affect vehicular traffic capacities.

b. On the same or similar map, demographic data, both resident and transient, in 1-mile annular increments, out to the sector boundaries defined in item 5 above and for each such sector. Population levels projected as peak values during the expected life of the plant should be used. If this information is incorporated elsewhere in the SAR, a specific reference thereto is suitable.

c. If means other than the use of private automobiles are assumed for any of the evacuation time estimates of item 5 above, these should be specified.

7. The identity and locations of the agency or agencies that would be responsible for providing warning and direction to offsite persons.

With respect to paragraph E, the PSAR should identify offsite hospital facilities that are expected to be able to provide (a) emergency care and (b) definitive patient care for acute radiation injury. Evidence should be given that preliminary contact with a relatively nearby hospital has established a willingness and potential capability to receive and treat individuals from the plant site who may have been affected by radiological emergencies.

With respect to paragraph F, the required descriptions of training planned for employees may incorporate by reference other sections of the PSAR as applicable and appropriate, for example, in the areas of radiation protection and firefighting. The PSAR should also include commitments to provide site familiarity training for others, not employed at the site,

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*If the proposed LPZ boundary is less than 5 miles from the plant, prior consultation with the NRC staff is suggested before submitting the analysis requested in this section.

who may require such familiarity to discharge their responsibilities in an emergency and a description of the training that ensures qualified medical services in emergency situations.

Although applicable primarily to the FSAR stage, Annex A of Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," particularly Sections 5, 6, and 8.1.1, should be consulted for guidance at the PSAR stage.*

13.3.2 Emergency Plan (FSAR)

At the FSAR stage, a comprehensive emergency plan should be submitted. This plan should be a physically separate document identified as Section 13.3 of the FSAR. The plan should show how the objectives and requirements of Parts I and III and paragraphs A to J of Part IV of Appendix E to 10 CFR Part 50 are to be implemented. Regulatory Guide 1.101* should be consulted. The information requirements identified in paragraphs 1, 6.a, and 6.b above for the PSAR should be provided in an appendix to the emergency plan, including any changes that may be necessary to update information previously submitted.

13.4 Review and Audit

The SAR should describe the applicant's plans for conducting reviews and audits of operating phase activities that are important to safety. The primary focus of attention should be:

1. On the procedures that will implement the licensee's responsibility pursuant to §50.59 of 10 CFR Part 50 relating to proposed changes, tests, and experiments, and

2. On the procedures for after-the-fact review and evaluation of unplanned events. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," contains guidance on conducting reviews and audits.

The PSAR should specifically indicate a commitment to meet this guidance or describe alternative means for meeting the same objectives. The FSAR should describe the applicant's detailed plans for conducting reviews and audits of operating phase activities that are important to safety.

13.4.1 Onsite Review (FSAR)

This section should specifically describe how the onsite organization functions with respect:

*The specific revision number and date of Regulatory Guide 1.101 used by the applicant should be specified.

1. To review of proposed changes to systems or procedures, tests, and experiments, and

2. To unplanned events that have operational safety significance.

The description should indicate how qualified members of the onsite operating organization will participate in the review of operating activities, either as part of their individual job responsibilities or as members of a functional review organization, to assist the plant manager.

13.4.2 Independent Review (FSAR)

This section should provide a detailed description of the provisions for performance of independent reviews of operating activities. Information in this section should describe the organizational method, composition and qualifications of the group, subjects to be reviewed, and the time such program is to be implemented relative to fuel loading of the (first) unit.

13.4.3 Audit Program (FSAR)

This section should provide a detailed description of the procedures and organization employed to implement the audit program with respect to operating activities and to verify compliance with the administrative controls and the quality assurance program.

13.5 Plant Procedures

This section of the SAR should describe administrative and operating procedures that will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, the SAR is not expected to include detailed written procedures. The PSAR should provide preliminary schedules for their preparation, and the FSAR should provide a brief description of the nature and content of the procedures and a schedule for their preparation.

13.5.1 Administrative Procedures

13.5.1.1 Conformance with Regulatory Guide 1.33. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," contains guidance on facility administrative policies and procedures. The SAR should specifically indicate whether the applicable portions of Regulatory Guide 1.33 concerning plant procedures will be followed. If such guidance will not be followed, the SAR should describe specific alternative methods that will be used and the manner of implementing them.

13.5.1.2 Preparation of Procedures. The PSAR and FSAR should provide a schedule for the preparation of appropriate written administrative procedures (see Section 13.5.1.1). The FSAR should identify the persons (by position) who have the responsibility for writing procedures and the persons who must approve them before they are implemented.

13.5.1.3 Procedures (FSAR). A description of administrative procedures should be provided and should include:

1. Standing orders to operations shift supervisors and shift crews including:

   a. The reactor operator's authority and responsibilities,

   b. The senior operator's authority and responsibilities,

   c. 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."

2. Special orders of a transient or self-cancelling character.

3. Equipment control procedures.

4. Control of maintenance and modifications.

5. Master surveillance testing schedule.

6. Procedures for logbook usage and control.

7. Temporary procedures.

13.5.2 Operating and Maintenance Procedures (FSAR)

13.5.2.1 Control Room Operating Procedures. This section should describe primarily the procedures that 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.

13.5.2.2 Other Procedures. 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 categories of procedures listed below should be included. If their general objectives and character are described elsewhere in the FSAR or the application, they may be described by specific reference thereto.

1. Plant radiation protection procedures.
2. Emergency preparedness procedures.
3. Instrument calibration and test procedures.
4. Chemical-radiochemical control procedures.
5. Radioactive waste management procedures.
6. Maintenance and modification procedures.
7. Material control procedures.
8. Plant security procedures.

13.6 Industrial Security

This section of the SAR should note that the applicant's plans for physical protection of the facility are described in a separate part of the application withheld from public disclosure pursuant to §2.790(d), 10 CFR Part 2, "Rules of Practice." Detailed security measures for the physical protection of nuclear power plants are required by §50.34(c), of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and applicable sections of 10 CFR Part 73, "Physical Protection of Plants and Materials." The regulatory position is set forth in Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," and includes an endorsement of ANSI Standard N18.17-1973, "Industrial Security for Nuclear Power Plants."

13.6.1 Preliminary Planning (PSAR)

At the time of submittal of the PSAR, the applicant's separate submittal should describe plans for the screening of personnel who are to be employed to work at the proposed plant, including personnel selection policies, employee performance and evaluation procedures, and the industrial security training program to be used to ensure that reliable and emotionally stable personnel are selected, maintained, and assigned to the plant staff and to the plant security force.

It should also describe plans for incorporating physical protection objectives and criteria into the design of the plant and the layout of equipment, including the following specific information as to how such plans will be or have been implemented:

1. Provide figures and/or drawings which identify the following:

a. Owner-controlled area, including private property markers, parking lot(s), and roads to be used for surveillance.

b. Protected area(s), including the associated isolation zone (clear area), physical barriers, access control points, lighting, intrusion monitoring and/or perimeter alarm systems, and roads or pathways to be used for surveillance.

c. Vital equipment and vital areas, including all access points.

d. Alarm station locations.

2. Describe the physical barrier construction for the protected and vital areas, and indicate the extent to which the positions set forth in ANSI N18.17-1973, Sections 3.3 and 3.4, are satisfied.

3. Describe the design features to be used for protecting all potential access points into the vital areas against unauthorized intrusion. Such features should include locking devices and intrusion detection devices.

4. Describe all intrusion alarms, emergency exit alarms, alarm systems, and line supervisory systems, and indicate the extent to which the level of performance and reliability specified by the Interim Federal Specification W-A-00450B (GSA-FSS), dated February 16, 1973, is met.

5. Describe the physical security provisions to be utilized in the design for the protection of security system service panels and wiring for protective devices, security communications systems, and door lock actuators.

6. Designate the person or group with the responsibility to conceive and detail security provisions in the physical plant design. If this responsibility is outside the owner organization, also specify the position within your organization responsible for the systematic review and control of the contracted activities.

13.6.2 Security Plan (FSAR)

At the time of submittal of the FSAR, the applicant's separate submittal should be a comprehensive description of the physical security program for the plant site. The information should include a description of the organization for security, a listing by title of all procedures to be established for plant security, 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, communication systems for security, provisions for maintenance and testing of security systems, and arrangements with law enforcement authorities for assistance in responding to security threats. The implementation schedule for the physical security program should be provided, including phases for multi-unit plants, where applicable.

Specific information for which guidance may be found in applicable referenced sections of ANSI N18.17-1973 and which should be included in the separate description is as follows:

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 areas and 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 that may present special vulnerability problems,

   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, describe in detail the extent to which the position of Section 3.2 of ANSI N18.17-1973 will be met.

3. The response capabilities of local law enforcement agencies should be fully described (Section 4.4 of ANSI N18.17-1973), 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.

4. A description should be included of any provisions for alternative interim protective measures during periods when one or more components of the total security system are not functioning.

14. INITIAL TEST PROGRAM

This chapter of the Safety Analysis Report should provide information on the initial test program for structures, systems, components, and design features for both the nuclear portion of the plant and the balance of the plant. The information provided should address major phases of the test program, including preoperational tests, initial fuel loading and initial criticality, low-power tests, and power-ascension tests. The Preliminary Safety Analysis Report (PSAR) should describe the scope of the applicant's initial test program. The PSAR should also describe the applicant's general plans for accomplishing the test program in sufficient detail to show that due consideration has been given to matters that normally require advance planning. The Final Safety Analysis Report (FSAR) should describe the technical aspects of the initial test program in sufficient detail to show that the test program will adequately verify the functional requirements of plant structures, systems, and components and that the sequence of testing is such that the safety of the plant will not be dependent on untested structures, systems, or components. The FSAR should also describe measures which ensure that (1) the initial test program will be accomplished with adequate numbers of qualified personnel, (2) adequate administrative controls will be established to govern the initial test program, (3) the test program will be used, to the extent practicable, to train and familiarize the plant operating and technical staff in the operation of the facility, and (4) the adequacy of plant operating and emergency procedures will be verified, to the extent practicable, during the period of the initial test program.

14.1 Specific Information To Be Included In Preliminary Safety Analysis Reports

14.1.1 Scope of Test Program

The major phases of the initial test program should be described and the overall test objectives and general prerequisites for each major phase should be discussed.

The PSAR should describe how the initial test program will be applied to the nuclear portion as well as the balance-of-plant portion of the facility. The organizations, including those of the applicant, that will participate in the development and execution of the test program and the general responsibilities of these organizations should be described. The PSAR should describe the applicant's planned involvement in the development and approval of test procedures, conduct of the tests, and review and approval of test results. The applicant's plans for having responsible design organizations participate in establishing test performance requirements and acceptance criteria should be described along with the applicant's plans for contracting the work of planning, developing, or conducting portions of the test program. The method by which the applicant will retain responsibility for and maintain control of such contracted work should be discussed.

14.1.2 Plant Design Features That Are Special, Unique, or First of a Kind

A summary description of preoperational and/or startup testing planned for each unique or first-of-a-kind principal design feature should be included in the PSAR. The summary test descriptions should include the test method and test objectives.

14.1.3 Regulatory Guides

The PSAR should describe the applicant's plans for using guidance in applicable regulatory guides in the development and conduct of the initial test program. An example of such guidance is Regulatory Guide 1.68, "Initial Test Program for Water-Cooled Reactor Power Plants." If such guidance will not be followed, the PSAR should describe specific alternative methods along with a justification for their use.

14.1.4 Utilization of Plant Operating and Testing Experiences at Other Reactor Facilities

The PSAR should describe the applicant's plans for the utilization of available information on reactor plant operating experiences to establish where emphasis may be warranted in the test program. The schedule, relative to the fuel loading date, for conducting the study or implementing the program should be described.

14.1.5 Test Program Schedule

A summary description should be provided on the overall schedule, relative to the expected fuel loading date, for developing and conducting the major phases of the test program. Information provided should establish the scheduled time period for developing detailed test procedures and the scheduled time period for conducting the tests for each major phase. Information should be provided to establish the compatibility of the test program schedule with the schedules for hiring and training of the plant operating and technical staff and for development of plant operating and emergency procedures, or reference should be made to appropriate sections of Chapter 13 of the PSAR.

14.1.6 Trial Use of Plant Operating and Emergency Procedures

The applicant's plans pertaining to the trial use of plant operating and emergency procedures during the period of the initial test program should be described.

14.1.7 Augmenting Applicant's Staff During Test Program

The applicant's general plans for the assignments of additional personnel to supplement his plant operating and technical staff during each major phase of the test program should be described. The PSAR should provide a description of the general responsibilities of the various augmenting organizations, a summary of the interrelationships and interfaces of the various organizations that will participate in the test program, the general qualifications of participating organizations, and the approximate schedule, relative to the fuel loading date, for augmenting the applicant's staff.

14.2 Specific Information To Be Included in Final Safety Analyis Reports

14.2.1 Summary of Test Program and Objectives

Describe the major phases of the test program and the specific objectives to be achieved for each major phase.

14.2.2 Organization and Staffing

A description of the applicant's organizational units and any augmenting organizations or other personnel that will manage, supervise, or execute any phase of the test program should be provided. This description should discuss the authorities, responsibilities, and degree of participation of each identified organizational unit and principal participants. The FSAR should describe how, and to what extent, the applicant's plant operating and technical staff will participate in each major test phase. Information pertaining to the experience and qualification of supervisory personnel and other principal participants that will be responsible for management, development, or conduct of each test phase should be provided or referenced elsewhere in the FSAR.

14.2.3 Test Procedures

The system that will be used to develop, review, and approve individual test procedures should be described, including the organizational units or personnel that are involved and their responsibilities. The FSAR should describe how organizations responsible for the design of the facility will participate in the establishment of performance requirements and acceptance criteria for testing plant structures, systems, and components and how such design organizations will interface with other participants involved in the test program. The FSAR should also describe the format of individual test procedures.

14.2.4 Conduct of Test Program

A description of the administrative controls that will govern the conduct of each major phase of the test programs should be provided. A description of the specific administrative controls that will be used to ensure that necessary prerequisites are satisfied for each major phase and for individual tests should also be provided. The FSAR should describe the methods to be followed in initiating plant modifications or maintenance that are determined to be required by the test program. The description should include the methods that will be used to ensure retesting following such modifications or maintenance and the involvement of design organizations and the applicant in the review and approval of proposed plant modifications. The administrative controls pertaining to adherence to approved test procedures during the conduct of the test program and the methods for effecting changes to approved test procedures should be described.

14.2.5 Review, Evaluation, and Approval of Test Results

The measures to be established for the review, evaluation, and approval of test results for each major phase of the program should be described. The specific controls to be established to ensure notification of affected and responsible organizations or personnel when test acceptance criteria are not met and the controls established to resolve such matters should also be described. A discussion should be provided on the applicant's plans pertaining to (1) approval of test data for each major test phase before proceeding to the next test phase and (2) approval of test data at each power test plateau (during the power-ascension phase) before increasing power level.

14.2.6 Test Records

The applicant's requirements pertaining to the disposition of test procedures and test data following completion of the test program should be described.

14.2.7 Conformance of Test Programs with Regulatory Guides

The applicant should list all those regulatory guides applicable to initial test programs that he plans to use for his test program. If such guidance will not be followed, the FSAR should describe specific alternative methods along with justification for their use.

14.2.8 Utilization of Reactor Operating and Testing Experiences in Development of Test Program

Information on the applicant's program for utilizing available information on reactor operating experiences in the development of his initial test program should be described, including the status of the program. The organizations participating in the program should be identified, their roles in the program discussed, and a summary description of their qualifications provided. The sources and types of information reviewed, the conclusions or findings, and the effect of the program on the initial test program should be described.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The schedule for development of plant procedures should be provided as well as a description of how, and to what extent, the plant operating and emergency procedures will be use-tested during the initial test program.

14.2.10 Initial Fuel Loading and Initial Criticality

The FSAR should describe the procedures that will guide initial fuel loading and initial criticality, including the safety and precautionary measures to be established for safe operation.

14.2.11 Test Program Schedule

The schedule, relative to the fuel loading date, for conducting each major phase of the test program should be provided. If the schedule will overlap initial test program schedules for other reactors at the site, a discussion should be provided on the effects of such schedule overlaps on organizations and personnel participating in the initial test program. The sequential test schedule for testing individual plant structures, systems, and components should be provided. Each test required to be completed before initial fuel loading should be identified.

The schedule for the development of test procedures for each major phase of the initial test program, including the time that will be available for review by NRC field inspectors of approved procedures, prior to their use, should be discussed.

14.2.12 Individual Test Descriptions

Test abstracts for each individual test that will be conducted during the initial test program should be provided. Emphasis should be placed on system and design features that (1) are relied on for the safe shutdown and cooldown of the facility under normal and faulted conditions, (2) are relied on for establishing conformance with limits or limiting conditions for operation that will be established by the technical specifications, and (3) are relied on to prevent or to limit or mitigate the consequences of anticipated transients and postulated accidents. The abstracts should identify each test by title, specify the prerequisites and major plant operating conditions necessary for each test (such as power level and mode of operation of major control systems), provide a summary description of the test method, describe the test objectives, and provide a summary of the acceptance criteria for each test.

15. ACCIDENT ANALYSES

The evaluation of the safety of a nuclear power plant should include analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the Commission's construction permit and operating license reviews of plants.

In previous chapters of the SAR, the structures, systems, and components important to safety should have been evaluated for their susceptibility to malfunctions and failures. In this chapter, the effects of anticipated process disturbances and postulated component failures should be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The situations analyzed should include anticipated operational occurrences (e.g., a loss of electrical load resulting from a line fault), off-design transients that induce fuel failures above those expected from normal operational occurrences, and postulated accidents of low probability (e.g., the sudden loss of integrity of a major component). The analyses should include an assessment of the consequences of an assumed fission product release that would result in potential hazards not exceeded by those from any accident considered credible.

Transient and Accident Classification

The approach outlined below is intended to organize the transients and accidents considered by the applicant and presented in the SAR in a manner that will:

1. Ensure that a sufficiently broad spectrum of initiating events has been considered,

2. Categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed, and

3. Permit the consistent application of specific acceptance criteria for each postulated initiating event.

To accomplish these goals, a number of disturbances of process variables and malfunctions or failures of equipment should be postulated. Each postulated initiating event should be assigned to one of the following categories:

1. Increase in heat removal by the secondary system (turbine plant),

2. Decrease in heat removal by the secondary system (turbine plant),

3. Decrease in reactor coolant system flow rate,

4. Reactivity and power distribution anomalies,

5. Increase in reactor coolant inventory,

6. Decrease in reactor coolant inventory,

7. Radioactive release from a subsystem or component, or

8. Anticipated transients without scram.

Typical initiating events that are representative of those that should be considered by the applicant in this chapter of the SAR are presented in Table 15-1. The evaluation of each initiating event should be presented in a separate subsection corresponding to the eight categories defined above. The information to be presented in these subsections is outlined in Section 15.X.X.

One of the items of information that should be discussed for each initiating event relates to its expected frequency of occurrence. Each initiating event within the eight major groups should be assigned to one of the following frequency groups:

1. Incidents of moderate frequency,

2. Infrequent incidents, or

3. Limiting faults.

The initiating events for each combination of category and frequency group should be evaluated to identify the events that would be limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed by the applicant. In some cases a qualitative comparison of similar initiating events may be sufficient to identify the specific initiating event that leads to the most limiting consequences. Only that initiating event should then be analyzed in detail.

It should be noted, however, that different initiating events in the same category/frequency group may be limiting when the multiplicity of consequences are considered. For example, within a given category/frequency group combination, one initiating event might result in the highest reactor coolant pressure boundary (RCPB) pressure while another initiating event might lead to minimum core thermal-hydraulic margins or maximum offsite doses.

Plant Characteristics Considered in the Safety Evaluation

A summary of plant parameters considered in the safety evaluation should be given; e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth and control rod insertion characteristics. A range of values should be specified for plant parameters that vary with fuel exposure or core reload. The range should be sufficiently broad to cover all expected changes predicted for the entire life of the plant. The permitted operating band (permitted fluctuations in a given parameter and associated uncertainties) on reactor system parameters should be specified. The most adverse conditions within the operating band should be used as initial conditions for transient analysis.

Assumed Protection System Actions

Settings of all protection system functions that are used in the safety evaluation should be listed. Typical protection system functions are reactor trips, isolation valve closures, ECCS initiation, etc. The uncertainty (combined effect of calibration error, drift, instrument error, etc.) associated with each function should also be listed together with the expected and maximum delay times.

15.X Evaluation of Individual Initiating Events

The applicant should provide an evaluation of each initiating event using the format of Section 15.X.X (e.g., 15.2.7 for a loss of normal feedwater flow initiating event). As shown in Table 15-1, a particular initiating event may be applicable to more than one category. The SAR sections should be appropriately referenced to indicate this.

The detailed information listed in Section 15.X.X, paragraphs 1 and 2, should be given for each initiating event. However, the extent of the quantitiative information in Section 15.X.X, paragraphs 3 through 5, that should be included will differ for the various initiating events. For those situations where a particular initiating event is not limiting, only the qualitative reasoning that led to that conclusion need be presented, along with a reference to the section that presents the evaluation of the more limiting initiating event. Further, for those initiating events that require a quantitative analysis, such an analysis may not be necessary for each of Section 15.X.X, paragraphs 3 through 5. For example, there are a number of plant transient initiating events that result in minimal radiological consequences. The applicant should merely present a qualitative evaluation to show this to be the case. A detailed evaluation of the radiological consequences need not be performed for each such initiating event.

15.X.X Event Evaluation

1. Identification of causes and frequency classification. For each event evaluated, include a description of the occurrences that lead to the initiating event under consideration. The probability of the initiating event should be estimated and the initiating event should be assigned to one of the following groups:

a. Incidents of moderate frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.

b. Infrequent incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.

c. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

2. Sequence of events and systems operation. The following should be discussed for each initiating event:

a. The step-by-step sequence of events from event initiation to the final stabilized condition. This listing should identify each significant occurrence on a time scale, e.g., flux monitor trip, insertion of control rods begin, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal initiated, and containment isolated. All required operator actions should also be identified.

b. The extent to which normally operating plant instrumentation and controls are assumed to function.

c. The extent to which plant and reactor protection systems are required to function.

d. The credit taken for the functioning of normally operating plant systems.

e. The operation of engineered safety systems that is required.

The effect of single failures in each of the above areas and the effect of operator errors should be discussed and evaluated. The discussion should provide enough detail to permit an independent evaluation of the adequacy of the system as related to the event under study. One method of systematically investigating single failures is the use of a plant operational analysis or a failure mode and effects analysis. The results of these types of analyses can be used to demonstrate that the safety actions required to mitigate the consequences of an event are provided by the safety systems essential to performing each safety action. A sample format is described in Transactions of the American Nuclear Society 1973 Winter Meeting (November 11-15, 1973, pp. 339-340).

3. Core and system performance.

a. Mathematical model. The mathematical model employed, including any simplifications or approximations introduced to perform the analyses, should be discussed. Any digital computer programs or analog simulations used in the analyses should be identified. If a set of codes is used, the method combining these codes should be described. Important output of each code should be presented and discussed under "results." Principal emphasis should be placed on the input data and the extent or range of variables investigated. This information should include figures showing the analytical model, flow path identification, actual computer listing, and complete listing of input data. The detailed description of mathematical models and digital computer programs or listings are preferably included by reference to documents available to the NRC with only summaries provided in the SAR text.

b. Input parameters and initial conditions. The input parameters and initial conditions used in the analyses should be clearly identified. Table 15-2 provides a representative list of these items. However, the initial values of other variables and additional parameters should be included in the SAR if they are used in the analyses of the particular event being analyzed.

The parameters and initial conditions used in the analyses should be suitably conservative for the event being evaluated except that anticipated transient without scram (ATWS) analyses should use realistic initial values. The bases used to select the numerical values that are input parameters to the analysis, including the degree of conservatism, should be discussed in the SAR.

c. Results. The results of the analyses should be presented and described in detail in the SAR. Key parameters should be presented as a function of time during the course of the transient or accident. The following are examples of parameters that should be included:

(1) Neutron power,

(2) Thermal power,

(3) Heat fluxes, average and maximum,

(4) Reactor coolant system pressure,

(5) Minimum CHFR, DNBR, or CPR, as applicable,

(6) Core and recirculation loop coolant flow rates (BWRs),

(7) Coolant conditions - inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam volume fractions,

(8) Temperatures - maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy,

(9) Reactor coolant inventory - total inventory and coolant level in various locations in the reactor coolant system,

(10) Secondary (power conversion) system parameters - steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, steam generator inventory, and

(11) ECCS flow rates and pressure differentials across the core, as applicable.

The discussion of results should emphasize the margins between the predicted values of various core parameters and the values of these parameters that would represent minimum acceptable conditions.

4. Barrier performance. This section of the SAR should discuss the evaluation of the parameters that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public.

a. Mathematical model. The mathematical model employed, including any simplifications or approximations introduced to perform the analyses, should be discussed. If the model is identical, or nearly identical, with that used to evaluate core performance, this should be stated in the SAR. In that case, only the differences, if any, between the models need be described.

A detailed description of the model used to evaluate barrier performance should be presented if it is significantly different from the core performance model. The information that should be included is indicated in paragraph 3 of Section 15.X.X, item a.

b. Input parameters and initial conditions. Any input parameters and initial conditions of variables relevant to the evaluation of barrier performance that were not presented and discussed in paragraph 3 of Section 15.X.X., item b, should be discussed in this section. The discussion should present the numerical values of the input to the analyses and should discuss the degree of conservatism of the selected values.

c. Results. The results of the analyses should be presented and described in detail in the SAR. As a minimum, the following information should be presented as a function of time during the course of the transient or accident:

(1) Reactor coolant system pressure,

(2) Steam line pressure,

(3) Containment pressure,

(4) Relief and/or safety valve flow rate,

(5) Flow rate from the reactor coolant system to the containment system, if applicable.

5. Radiological consequences. This section of the SAR should summarize the assumptions, parameters, and calculational methods used to determine the doses that result from limiting faults and infrequent incidents. Sufficient information should be given in this section to fully substantiate the results and to allow an independent analysis to be performed by the NRC staff. Thus, this section should include all of the pertinent plant parameters that are required to calculate doses for the exclusion boundary and the low population zone as well as those locations within the exclusion boundary where significant site-related activities may occur (e.g., the control room).

The elements of the dose analysis that are applicable to several accident types or that are used many times throughout Chapter 15 can be summarized in this section (or cross-referred) with the bulk of the information appearing in appendices.

If there are no radiological consequences associated with a given initiating event, this section for the event should simply contain a statement indicating that containment of the activity was maintained and by what margin.

An analysis should be provided for each limiting event. These analyses should be based on design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 CFR Part 100 criteria. These design basis assumptions can, for the most part, be found in regulatory guides that deal with radiological releases. For instance, when calculating the radiological consequences of a loss-of-coolant accident (LOCA), it is suggested that the assumptions given in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," be used. This analysis should be referred to as the "design basis analysis." There may be instances in which the applicant will not agree with the conservative margins inherent in the design basis approach approved by the NRC staff or the applicant may desire to provide a "realistic analysis" for comparison purposes. If this is the case, the applicant may provide an indication of the assumptions he believes to be adequately conservative, but the known NRC assumptions should nevertheless be used in the design basis analysis.

Any "realistic analysis" provided will help quantify the margins that are inherent in the design basis approach. A "realistic analysis" need not include a consequence assessment and may be limited to a presentation of assumptions that are more likely to be obtained than those used for purposes of design.

The parameters and assumptions used for these analyses, as well as the results, should be presented in tabular form. Table 15-3 provides a representative list of these items. Table 15-4 summarizes additional items that should be provided when dealing with specific types of accidents. When possible, the summary tabulation should provide the necessary quantitative information. If, however, a particular assumption cannot be simply or clearly stated in the table, the table should reference a section or an appendix that adequately discusses the information.

Judgment should be used in eliminating unnecessary parameters from the summary table or in adding parameters of significance that do not appear in Table 15-3 or 15-4. The summary table should have two columns. One column should indicate the assumptions used in the design basis analysis, while the other should indicate assumptions used in the realistic analysis.

A diagram of the dose computation model, labeled "Containment Leakage Dose Model," should be appended to Chapter 15. An explanation of the model should accompany the diagram. The purpose of the appendix is to clearly illustrate the containment modeling, the leakage or transport of radioactivity from one compartment to another or to the environment, and the presence of engineered safety features (ESF) such as filters or sprays that are called on to mitigate the consequences of the LOCA. The diagram should employ easily identifiable symbols, e.g., squares to represent the containment or various portions of it, lines with arrowheads drawn from one compartment to another or to the environment to indicate leakage or transport of radioactivity, and other suitably labeled or defined symbols to indicate the presence of ESF filters or sprays. Individual sketches (or equivalent) may be used for each significant time interval in the containment leakage history (e.g., separate sketches showing the pulldown of a dual containment annulus and the exhaust and recirculation phases once negative pressure in the annulus is achieved, with the appropriate time intervals given).

In presenting the assumptions and methodology used in determining the radiological consequences, care should be taken to ensure that analyses are adequately supported with backup information, either by reporting the information where appropriate, by referencing other sections within the SAR, or by referencing documents readily available to the NRC staff. Such information should include the following:

a. A description of the mathematical or physical model employed, including any simplifications or approximations introduced to perform the analyses.

b. An identification and description of any digital computer program or analog simulation used in the analysis. The detailed description of mathematical models and programs are preferably included by reference with only summaries provided in the SAR text.

c. An identification of the time-dependent characteristics, activity, and release rate of the fission products or other transmissible radioactive materials within the containment system that could escape to the environment via leakages in the containment boundaries and leakage through lines that could exhaust to the environment.

d. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

e. A discussion of the extent of system interdependency (containment system and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the containment system or other sources (e.g., from spent fuel handling areas), such as the contribution of (1) containment water spray systems, (2) containment air cooling systems, (3) air purification and cleanup systems, (4) reactor core spray or safety injection systems, (5) postaccident heat removal systems, and (6) main steam line isolation valve leakage control systems (BWR).

This section should present the results of the dose calculations giving the potential 2-hour integrated whole body and thyroid doses for the exclusion boundary. Similarly, it should provide the doses for the course of the accident at the closest boundary of the low population zone (LPZ) and, when significant, the doses to the control room operators during the course of the accident. Other organ doses should be presented for those cases where a release of solid fission products or transuranic elements are postulated to be released to the containment atmosphere.

TABLE 15-1

REPRESENTATIVE INITIATING EVENTS TO BE ANALYZED IN SECTIONS 15.X.X OF THE SAR

1. Increase in Heat Removal by the Secondary System

    1.1 Feedwater system malfunctions that result in a decrease in feedwater temperature

    1.2 Feedwater system malfunctions that result in an increase in feedwater flow

    1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow

    1.4 Inadvertent opening of a steam generator relief or safety valve

    1.5 Spectrum of steam system piping failures inside and outside of containment in a PWR

2. Decrease in Heat Removal by the Secondary System

    2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow

    2.2 Loss of external electric load

    2.3 Turbine trip (stop valve closure)

    2.4 Inadvertent closure of main steam isolation valves

    2.5 Loss of condenser vacuum

    2.6 Coincident loss of onsite and external (offsite) a.c. power to the station

    2.7 Loss of normal feedwater flow

    2.8 Feedwater piping break

3. Decrease in Reactor Coolant System Flow Rate

    3.1 Single and multiple reactor coolant pump trips

    3.2 BWR recirculation loop controller malfunctions that result in decreasing flow rate

    3.3 Reactor coolant pump shaft seizure

TABLE 15-1 (Continued)

3.4 Reactor coolant pump shaft break

4. Reactivity and Power Distribution Anomalies

4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling

4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power)

4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods

4.4 Startup of an inactive reactor coolant loop or recirculating loop at an incorrect temperature

4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate

4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR

4.7 Inadvertent loading and operation of a fuel assembly in an improper position

4.8 Spectrum of rod ejection accidents in a PWR

4.9 Spectrum of rod drop accidents in a BWR

5. Increase in Reactor Coolant Inventory

5.1 Inadvertent operation of ECCS during power operation

5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory

5.3 A number of BWR transients, including items 2.1 through 2.6 and item 1.2.

6. Decrease in Reactor Coolant Inventory

6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR

TABLE 15-1 (Continued)

6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment

6.3 Steam generator tube failure

6.4 Spectrum of BWR steam system piping failures outside of containment

6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR

6.6 A number of BWR transients, including items 2.7, 2.8, and 1.3

7. Radioactive Release from a Subsystem or Component

7.1 Radioactive gas waste system leak or failure

7.2 Radioactive liquid waste system leak or failure

7.3 Postulated radioactive releases due to liquid tank failures

7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings

7.5 Spent fuel cask drop accidents

8. Anticipated Transients Without Scram

8.1 Inadvertent control rod withdrawal

8.2 Loss of feedwater

8.3 Loss of a.c. power

8.4 Loss of electrical load

8.5 Loss of condenser vacuum

8.6 Turbine trip

8.7 Closure of main steam line isolation valves

TABLE 15-2

INPUT PARAMETERS AND
INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS

Neutron Power

Moderator Temperature Coefficient of Reactivity

Moderator Void Coefficient of Reactivity

Doppler Coefficient of Reactivity

Effective Neutron Lifetime

Delayed Neutron Fraction

Average Heat Flux

Maximum Heat Flux

Minimum DNBR, CHFR, or CPR

Axial Power Distribution

Radial Power Distribution

Core Coolant Flow Rate

Recirculation Loop Flow Rate (BWR)

Core Coolant Inlet Temperature

Core Average Coolant Temperature (PWR)

Core Average Steam Volume Fraction (BWR)

Core Coolant Average Exit Temperature, Steam Quality, and Steam Void Fraction

Hot Channel Coolant Exit Temperature, Steam Quality, and Steam Void Fraction

Maximum Fuel Centerline Temperature

Reactor Coolant System Inventory (lb)

Coolant Level in Reactor Vessel (BWR)

Coolant Level in Pressurizer (PWR)

TABLE 15-2 (Continued)

Reactor Coolant Pressure

Steam Flow Rate

Steam Pressure

Steam Quality (temperature if superheated)

Feedwater Flow Rate

Feedwater Temperature

CVCS Flow and Boron Concentration (if these vary during the course of the transient or accident being analyzed)

Control Rod Worth, Differential, and Total

TABLE 15-3

REPRESENTATIVE PARAMETERS TO BE TABULATED* FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents		
a. Stretch power level		
b. Burnup		
c. Percent of fuel perforated		
d. Release of activity by nuclide		
e. Iodine fractions (organic, elemental, and particulate)		
f. Reactor coolant activity before the accident (and secondary coolant activity for PWR). Two values for primary system iodine activity concentration should be given. These two values should indicate (1) the maximum allowable equilibrium iodine concentration and (2) the maximum allowable concentration resulting from a preaccident iodine spike.		
2. Data and assumptions used to estimate activity released		
a. Primary containment volume and leak rate		
b. Secondary containment volume and leak rate		
c. Valve movement times		
d. Adsorption and filtration efficiencies		
e. Recirculation system parameters (flow rates versus time, mixing factor, etc.)		
f. Containment spray first order removal lambdas as determined in Section 6.2.3		
g. Containment volumes		
h. All other pertinent data and assumptions		

*As applicable to the event being described.

TABLE 15-3 (Continued)

	Design Basis Assumptions	Realistic Assumptions
3. Dispersion Data		
a. Location of points of release		
b. Distances to applicable receptors (e.g., control room, exclusion boundary, and LPZ)		
c. χ/Qs at control room, exclusion boundary, and LPZ (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days)		
4. Dose Data		
a. Method of dose calculation		
b. Dose conversion assumptions		
c. Peak [or f(t)] concentrations in containment		
d. Doses (whole body and thyroid doses for LPZ and exclusion boundary; beta, gamma, and thyroid doses for the control room)		

TABLE 15-4

ADDITIONAL PARAMETERS AND INFORMATION TO BE PROVIDED OR REFERENCED IN THE SUMMARY TABULATION FOR SPECIFIC DESIGN BASIS ACCIDENTS

1. Loss-of-Coolant Accident (Section 15.6.5)

   a. Hydrogen Purge Analysis

      (1) Holdup time prior to purge initiation (assuming recombiners are inoperative)

      (2) Iodine reduction factor

      (3) χ/Q values at appropriate time of release

      (4) Purge rates for at least 30 days after initiation of purge

      (5) LOCA plus purge dose at LPZ

   b. Equipment Leakage Contribution to LOCA Dose

      (1) Iodine concentration in sump water after LOCA

      (2) Maximum operational leak rate through pump seals, flanges, valves, etc.

      (3) Maximum leakage assuming failure and subsequent isolation of a component seal

      (4) Total leakage quantities for (2) and (3)

      (5) Temperature of sump water vs time

      (6) Time intervals for automatic and operator action

      (7) Leak paths from point of seal or valve leakage to the environment

      (8) Iodine partition factor for sump water vs temperature of water

      (9) Charcoal adsorber efficiency assumed for iodine removal

   c. Main Steam Line Isolation Valve Leakage Control System Contribution to LOCA Dose (BWR)

      (1) Time of system actuation

      (2) Fraction of isolation valve leakage from each release point

TABLE 15-4 (Continued)

(3) Flow rates vs time for each release path

(4) Location of each release point

(5) Transport time to each release point

2. Waste Gas System Failure (Section 15.7.1)

a. Activity transfer times to waste gas system components

b. Number of tanks or other holdup components

c. Tank volumes

d. Charcoal bed delay times for Xe and Kr

e. Seismic classification of tank and associated piping

f. Decontamination factors of components

g. Primary coolant volume

h. Isotopic activity in each system component including daughter products

i. Time to isolate air ejector

j. Delay time in delay pipe

k. Design basis activity measured at air ejector (Ci/sec) including contribution due to activity spiking in coolant

3. Main Steam Line and Steam Generator Tube Failures (Sections 15.1.5, 15.6.3, 15.6.4)

a. Characterization of primary and secondary (PWR) system. Sufficient information should be given, as appropriate, to adequately describe the time-histories from accident initiation until accident recovery is complete for temperatures, pressures, steam generator water capacity, steaming rates, feedwater rates, blowdown rates, and primary-to-secondary leakage rates.

b. Potential increase in iodine release rate above the equilibrium value (i.e., iodine spiking) from the fuel to the primary coolant as a result of the accident or a pre-accident primary system transient.

c. Chronological list of system response times, operator actions, valve closure times, etc.

TABLE 15-4 (Continued)

d. Steam and water release quantities and all assumptions made in their computation

e. Description of the iodine transport mechanism and release paths between the primary system and the environment. The bases for an assumed partitioning of iodine between liquid and steam phases should be described and justified.

f. Possible fuel rod failure resulting from the accident, assuming the most reactive control rod remains in its fully withdrawn position.

g. Possible steam generator tube failure resulting from a PWR steam line break accident.

4. Fuel Handling Accident (in the Containment and Spent Fuel Storage Buildings) (Section 15.7.4)

a. Number of fuel rods in core

b. Number, burnup, and decay time of fuel rods assumed to be damaged in the accident

c. Radial peaking factor for the rods assumed to be damaged

d. Earliest time after shutdown that fuel handling begins

e. Amounts of iodines and noble gases released into pool

f. Pool decontamination factors

g. Time required to automatically switch from normal containment purge operation to either safety-grade filters or isolation

h. Amount of radioactive release not routed through ESF-grade filters.

The following items should be provided to determine if the calculational methods of Regulatory Guide 1.25 (Safety Guide 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," apply:

i. Maximum fuel rod pressurization

j. Minimum water depth between top of fuel rods and fuel pool surface

TABLE 15-4 (Continued)

k. Peak linear power density for the highest power assembly discharged

l. Maximum centerline operating fuel temperature for the fuel assembly in item k above

m. Average burnup for the peak assembly in item k above

5. Control Rod Ejection and Control Rod Drop Accidents (Sections 15.4.8 and 15.4.9)

   a. Percent of fuel rods undergoing clad failure

   b. Radial peaking factors for rods undergoing clad failure

   c. Percent of fuel reaching or exceeding melting temperature

   d. Peaking factors for fuel reaching or exceeding melting temperature

   e. Percent of core fission products assumed released into reactor coolant

   f. Summary of primary and secondary system parameters used to determine the activity release through the secondary system (PWRs only). The information specified in items 3a, b, c, d, and e of this table should also be provided for this accident.

   g. Summary of containment system parameters used to determine activity release terms from containment leak paths

   h. Summary of system parameters and decontamination factors used to determine activity release from condenser leak paths (BWR)

6. Spent Fuel Cask Drop (Section 15.7.5)

   a. Number of fuel elements in largest capacity cask

   b. Number, burnup, and decay time of fuel elements in cask assumed to be damaged

   c. Number, burnup, and decay time of fuel elements in pool assumed to be damaged as a consequence of a cask drop (if any)

   d. Average radial peaking factor for the rods assumed to be damaged

TABLE 15-4 (Continued)

e. Earliest time after reactor fueling that cask loading operations begin

f. Amounts of iodines and noble gases released into air and into pool

g. Pool decontamination factors, if applicable

7. Failure of Small Lines Carrying Primary Coolant Outside Containment (Section 15.6.2)

a. Detailed description of primary system response, leakage rate, operator action, valve closure times, etc. Also, figures indicating primary system pressure and temperature and primary coolant leakage versus time for the duration of the accident should be provided, as well as chronological listing of system response times, operator actions, valve closure times, etc.

b. Summary of primary system iodine activity during the accident and its effect on the calculated accident consequences as described in item 3.b of this table

c. Iodine transport mechanism and release paths from the leak point to the environment

16. TECHNICAL SPECIFICATIONS

Section 50.36 of 10 CFR 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 on facility operation for, among other purposes, the protection of the health and safety of the public. Each applicant for an operating license is required to submit proposed Technical Specifications and their bases for the facility. They should be consistent with the content and format of the Standard Technical Specifications available from the Commission for the appropriate nuclear steam supply system (NSSS) vendor. After review and needed modification by the NRC staff, these Technical Specifications will be issued by the Commission as Appendix A to the operating license.

16.1 Preliminary Technical Specifications (PSAR)

An application for a construction permit should include preliminary Technical Specifications that identify and provide justification for the selection of variables, conditions, or other limitations that are determined to be probable subjects of the final Technical Specifications. Special attention should be given to those areas that influence the final design in order to minimize later facility modifications to accommodate conditions of the final Technical Specifications. In particular, this review should determine the design suitability of those features and specifications that affect the type, capacity, and number of safety-related systems and the capability for performance of surveillance activities involving those safety-related systems.

The preliminary Technical Specifications and bases should be included in this chapter of the PSAR. The submittal should be consistent with the format and content of the NRC Standard Technical Specification for the appropriate NSSS vendor. Each specification should be as complete as possible and should include preliminary numerical values, graphs, tables, and other data. References to the applicable sections of the PSAR that support the bases and provide clarifying details for each specification should be supplied. Justification should be provided for deletions from or additions to the Standard Technical Specifications pertinent to the selected NSSS vendor.

16.2 Proposed Final Technical Specifications (FSAR)

The Technical Specifications submitted in support of an operating license application should be the finalized version of those specifications originally included in the PSAR. The numerical values, graphs, tables, and other data should reflect the final refinements in design, results of tests or experiments, and expected method of operation. This information should be included in this chapter of the FSAR.

17. QUALITY ASSURANCE

In order to provide assurance that the design, construction, and operation of the proposed nuclear power plant are in conformance with applicable regulatory requirements and with the design bases specified in the license application, it is necessary that a quality assurance (QA) program be established by the applicant. In this chapter of the SAR, the applicant should provide a description of the QA program to be established and executed during the design, construction, preoperational testing, and operation of the nuclear power plant. The QA program must be established at the earliest practical time consistent with the schedule for accomplishing the activity. Where some portions of the QA program have not yet been established at the time the SAR is prepared because the activity will be performed in the future, the description should also provide a schedule for implementation. The program must meet the requirements of Appendix B to 10 CFR Part 50. The inspection and survey systems required by §50.55a, "Codes and Standards," of 10 CFR Part 50 may be used in partial fulfillment of these requirements to the extent that they are shown by the description of the QA program to satisfy the applicable requirements of Appendix B.

In order to facilitate the presentation of the information, the QA program for each of the major organizations involved in executing the QA program should include the information described (either separately for each organization or integrally for all organizations) in accordance with the outline presented below. It is not intended to dictate the format of any QA Program Manual; that is left to the discretion of the applicant. It is required, however, that the description in the SAR address, at a minimum, each of the criteria in Appendix B in sufficient detail to enable the reviewer to determine whether and how all the requirements of the appendix will be satisfied in accordance with §50.34 of 10 CFR Part 50. Reference to appropriate portions of other sections of the SAR may suffice.

NRC regulatory guides and the documents entitled "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," (WASH 1283), "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," (WASH 1309), and "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," (WASH 1284) contain guidance on acceptable methods of implementing portions of the quality assurance program.* The SAR should specifically indicate whether this guidance will be followed. If such guidance will not be followed, the SAR should describe specific alternative methods that will be used and the manner of implementing them and should identify the organizations responsible for their implementation.

* WASH 1283, 1284, and 1309 contain a number of draft standards. As these draft standards are issued as approved American National Standards, it is expected that they will be endorsed by regulatory guides. The applicability of the regulatory guide versus the draft standard will be addressed in the implementation section of each guide or in amendments to this Standard Format.

Where a portion of the QA program to be implemented will follow the guidance provided by a regulatory guide, WASH 1283, WASH 1309, or WASH 1284, the program description may consist of a statement that the guidance will be followed for that portion of the QA program. When these documents are used in describing the QA program, the applicant should indicate how the guidance documents will be applied to portions of the QA program and should delineate the organizational element responsible for implementing various provisions of the respective guidance documents within each major organization in the project, including that of the applicant, the architect-engineer, the nuclear steam system supplier, the constructor, the construction manager (if other than the constructor).

17.1 Quality Assurance During Design and Construction

17.1.1 Organization

17.1.1.1. The PSAR should describe clearly the authority and duties of persons and organizations performing the QA functions of assuring that the QA program is established and executed and of verifying that an activity has been correctly performed. The PSAR should provide organization charts and functional responsibility descriptions that denote the lines of responsibility and areas of authority within each of the major organizations in the project, including those of the applicant, the architect-engineer, the nuclear steam system supplier, the constructor, and the construction manager (if other than the constructor). These charts and descriptions should present the structure of QA organizations involved as well as other functional organizations performing activities affecting quality in design, procurement, manufacturing, construction and installation, testing, inspection, and auditing with clear delineation of their responsibility, authority, and relationship to corporate management. In addition, a single overall project organization chart should be included showing how the major organizations or companies working directly for the applicant on the project interrelate with one another.

17.1.1.2. The PSAR should describe the level of management responsible for establishing the QA policies, goals, and objectives and should describe the continuing involvement of this management level in QA matters. The PSAR should tell what position has overall authority and responsibility for the QA program and tell what position is responsible for final review and approval of the QA program and related manuals. The qualification requirements of the principal QA and quality control positions should be described.

17.1.1.3. The PSAR should describe those measures which assure that persons and organizations performing QA functions have sufficient authority and organizational freedom to (1) identify quality problems, (2) initiate, recommend, or provide solutions, and (3) verify implementation of solutions. The PSAR should describe the measures which assure that persons and organizations assigned the responsibility for checking, auditing, inspecting, or otherwise verifying that an activity has been correctly performed report to a management level such that this required authority and organizational freedom, including sufficient independence from the pressures of production, are provided. Irrespective of the organizational structure, the PSAR should describe how the individual or individuals with primary responsibility for assuring effective implementation of the QA program at any location where activities subject to the control of the QA program are being performed will have direct access to such levels of management as may be necessary to carry out this responsibility. The PSAR should indicate from whom the persons performing QA functions receive technical direction for performing QA tasks and administrative control (salary review, hire/fire, position assignment). The PSAR should identify those positions or organizations which have written delegated responsibility and authority to stop work or control further processing, delivery, installation, or use of nonconforming items until proper disposition of the deficiency has been approved.

The PSAR should describe how requirements will be imposed on contractors and subcontractors to assure that individuals or groups within their organizations performing QA functions have sufficient authority and organizational freedom to effectively implement their respective QA programs.

17.1.1.4. The PSAR should describe the extent to which the applicant will delegate to other contractors the work of establishing and executing the QA program or any part thereof. A clear delineation of those QA functions which are implemented within the applicant's QA organization and those which are delegated to other organizations should be provided in the PSAR. The PSAR should describe the method by which the applicant will retain responsibility for and maintain control over those portions of the QA program delegated to other organizations and should identify the organization responsible for verifying that delegated QA functions are properly carried out. The PSAR should identify major work interfaces for activities affecting quality and describe how clear and effective lines of communication exist between the applicant and his principal contractors to assure necessary coordination and control of the QA program.

17.1.2 Quality Assurance Program

17.1.2.1. The QA program in the PSAR should cover each of the criteria in Appendix B to 10 CFR Part 50 in sufficient detail to permit a determination as to whether and how all of the requirements of Appendix B will be satisfied. The PSAR should (1) describe the extent to which the QA program will conform to various provisions of WASH 1283, WASH 1309, and regulatory guides that provide guidance on acceptable methods of implementing portions of the QA program and (2) identify the organizational element responsible for implementing these provisions. If the applicant elects not to follow the above guidance, the PSAR should describe in detail equivalent to that furnished in the NRC guidance the alternative methods that will be used and the manner of implementing them and should indicate the organizations responsible for their implementation.

17.1.2.2. The PSAR should identify the safety-related structures, systems, and components to be controlled by the QA program.

17.1.2.3. The PSAR should describe the measures which assure that the QA program is being established at the earliest practicable time consistent with the schedule for accomplishing activities affecting quality for the project. That is, the PSAR should describe how the QA program is being established in advance of the activity to be controlled and how it will be implemented as the activity proceeds. Those activities affecting quality initiated prior to the submittal of the PSAR, such as establishing information required to be included in the PSAR, design and procurement, safety-related site testing and evaluation, and preparation activities should be identified in the PSAR. The PSAR should describe how these activities are controlled by a QA program which complies with Appendix B to 10 CFR Part 50.

17.1.2.4. The PSAR should describe how the QA program is documented by written policies, procedures, or instructions and how it will be implemented in accordance with these policies, procedures, or instructions. The PSAR should include a listing of QA program procedures or instructions that will be used to implement the QA program for each major activity such as design, procurement, construction, etc. The procedure list should identify which criteria of Appendix B to 10 CFR Part 50 are implemented by each procedure. In the event certain required procedures are not yet established, a schedule for their preparation should be provided in the PSAR.

17.1.2.5. The PSAR should summarize the corporate QA policies, goals, and objectives; and it should describe how disputes involving quality are resolved.

17.1.2.6. The PSAR should describe the program that provides adequate indoctrination and training of personnel performing activities affecting quality to assure that suitable proficiency is achieved and maintained. The PSAR should describe how the indoctrination and training program will assure that:

1. Personnel performing activities affecting quality are appropriately trained in the principles and techniques of the activity being performed,

2. Personnel performing activities affecting quality are instructed as to purpose, scope, and implementation of governing manuals, policies. and procedures,

3. Appropriate training procedures are established, and

4. Proficiency of personnel performing activities affecting quality is maintained.

17.1.2.7. The PSAR should describe the qualification requirements for the position or positions responsible for assuring effective implementation of the QA program of the applicant and of his major contractors.

17.1.2.8. The PSAR should describe the measures that assure that activities affecting quality will be accomplished under suitable controlled conditions, including (1) the use of appropriate equipment, (2) a suitable environment for accomplishing the activity, e.g., adequate cleanliness, and (3) compliance with necessary prerequisites for the given activity.

17.1.2.9. The PSAR should describe the measures that assure that there is regular management review of the QA program to assess its effectiveness and the adequacy of its scope, and implementation. The PSAR should describe the provisions for reviews by management above or outside the QA organization to assure achieving an objective program assessment.

The PSAR should describe the measures that assure that the QA organization of the applicant will (1) review and document agreement with the QA programs of the principal contractors and (2) conduct or have conducted audits of the contractors' QA program activities.

17.1.2.10. The PSAR should provide a summary description of advanced planning that demonstrates control of quality-related activities including management and technical interfaces between the contructor, the architect-engineer, the nuclear steam system supplier, and the applicant during the phaseout of design and construction and during preoperational testing and plant turnover.

17.1.2.11. The PSAR should describe provisions for maintaining the QA program description current.

17.1.3 Design Control

17.1.3.1. The PSAR should describe the design control measures that assure that (1) applicable regulatory requirements and design bases for safety-related structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions, (2) appropriate quality standards are specified in design documents, and (3) deviations from such standards are controlled.

17.1.3.2. The PSAR should describe measures that assure that adequate review and selection for application suitability is conducted for materials, parts, equipment, and processes that are essential to safety-related functions of the structures, systems, and components. The PSAR should describe provisions that assure that standard commercial or so-called "off the shelf" materials, parts, and equipment also receive adequate application review and selection.

17.1.3.3. The PSAR should describe the program for applying design control measures to such aspects of design as reactor physics; stress, thermal, hydraulic, and accident analysis; materials compatibility; and accessibility for maintenance, inservice inspection, and repair and should describe measures for delineation of acceptance criteria for inspections and tests.

17.1.3.4. The PSAR should describe measures that assure verification or checking of design adequacy, such as design reviews, use of alternative calculational methods, or performance of a qualification testing program under the most adverse design conditions. The PSAR should identify the positions or organizations responsible for design verification or checking and should describe measures that assure that the verifying or checking process is performed by individuals or groups other than those who performed the original design, but who may be from the same organization.

17.1.3.5. The PSAR should describe measures for identifying and controlling design interfaces, both internal and external, and for coordination between participating design organizations. The PSAR should describe measures in effect between participating design organizations for review, approval, release, distribution, collection, and storage of documents involving design interfaces and changes thereto. The PSAR should describe how these measures will assure that these design documents are controlled in a timely manner to prevent inadvertent use of superseded design information.

17.1.3.6. The PSAR should describe the measures that will be employed to assure that design changes, including field changes, are subject to the same design controls that were applied to the original design and are reviewed and approved by the organization that performed the original design unless the originating organization designates another responsible organization.

17.1.4 Procurement Document Control

17.1.4.1. The PSAR should describe measures that assure that documents, and changes thereto, for procurement of material, equipment, and services, whether purchased by the applicant or the contractors or subcontractors, correctly include or reference the following as necessary to achieve required quality:

1. Applicable regulatory, code, and design requirements,

2. Quality assurance program requirements,

3. Requirements for supplier documents such as instructions, procedures, drawings, specifications, inspection and test records, and supplier QA records to be prepared, submitted, or made available for purchaser review or approval,

4. Requirements for the retention, control, and maintenance of supplier QA records,

5. Provision for purchaser's right of access to suppliers' facilities and work documents for inspection and audit, and

6. Provision for supplier reporting and disposition of nonconformances from procurement requirements.

17.1.4.2. The PSAR should describe (1) measures that clearly delineate the control responsibilities and action sequence to be taken in the preparation, review, approval, and issuance by competent personnel of procurement documents and (2) measures that assure that changes or revisions of procurement documents are subject to the same review and approval requirements as the original documents.

17.1.4.3. The PSAR should describe measures that assure (1) that procurement documents require suppliers to have and implement a documented QA program for purchased materials, equipment, and services to an extent consistent with their importance to safety, (2) that the purchaser has evaluated the supplier before the award of the procurement order or contract to assure that the supplier can meet the procurement requirements, and (3) that procurement documents for spare or replacement items will be subject to controls at least equivalent to those used for the original equipment.

17.1.5 Instructions, Procedures, and Drawings

17.1.5.1. The PSAR should describe measures that assure that activities affecting quality such as design, procurement, manufacturing, construction and installation, testing, inspection, and auditing are prescribed by appropriately documented instructions, procedures, or drawings and that these activities will be conducted in accordance with these documents.

17.1.5.2. The PSAR should describe the system whereby the documented instructions, procedures, and drawings will include appropriate quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples and weld radiographic acceptance standards) acceptance criteria for determining that prescribed activities have been satisfactorily accomplished.

17.1.6 Document Control

17.1.6.1. The PSAR should describe those measures established to control the issuance of documents such as instructions, procedures, and drawings, including changes thereto, that prescribe all activities affecting quality. The description should cover control measures that assure that:

1. Documents are reviewed for adequacy (i.e., information is clearly and accurately stated) and are approved by authorized personnel for issuance and use at locations where the prescribed activity will be performed before the activity is started,

2. Means such as use of updated master document lists exist to assure that obsolete or superseded documents are replaced in a timely manner by updated applicable document revisions, and

3. Document changes are reviewed and approved by the same organizations that performed the original review and approval unless delegated by the originating organization to another responsible organization.

17.1.6.2. The PSAR should identify the types of documents to be controlled and the group responsible for review, approval, and issuance of documents and changes thereto.

17.1.7 Control of Purchased Material, Equipment, and Services

17.1.7.1. The PSAR should describe those measures that assure that material, equipment, and services purchased directly by the applicant or his contractors and subcontractors will conform to procurement document requirements. The PSAR should describe the measures that provide, as appropriate, for:

1. Evaluation and selection of sources of supply before the award of the procurement order or contract,

2. Surveillance at the supplier's facility by the purchaser or his representative in accordance with written procedures during design, manufacture, inspection, and test of the procured item or service to verify compliance with quality requirements,

3. Source and/or receipt inspection in accordance with written procedures and acceptance criteria of procured items furnished by the supplier,

4. Documentary evidence at the site from the supplier that procured items meet procurement quality requirements such as codes, standards, or specifications. The PSAR should describe measures established by the applicant to (a) examine and indicate acceptance of this documented evidence during source or receipt inspection and (b) assure that this documented evidence is available at the nuclear power plant site prior to installation or use of the procured item and that the documentation will be retained at the plant site, and

5. Periodic verification of supplier's certificates of conformance to assure that they are meaningful.

17.1.7.2. The PSAR should describe measures whereby the applicant or his designated representative will audit and evaluate the effectiveness of the control of quality-related activities of contractors and subcontractors at a frequency and extent consistent with the importance to safety, complexity, and quantity of the item or service being furnished.

17.1.8 Identification and Control of Materials, Parts, and Components

The PSAR should describe measures established to identify and control items such as materials, parts, and components, including partially fabricated assemblies, to prevent use of incorrect or defective items. The PSAR should describe measures that assure (1) that identification of the item, (i.e., heat number, part number, serial number, or other appropriate marking) is maintained either on the item or on records traceable to the item and verified, as required, throughout fabrication, erection, installation, and use of the item and (2) that the method and location of the identification does not affect the function or quality of the item being identified.

17.1.9 Control of Special Processes

The PSAR should describe measures established to control special processes such as welding, heat treating, nondestructive testing, and electrochemical machining and to assure that they are accomplished by qualified personnel using written procedures qualified in accordance with applicable codes, standards, specifications, or other special requirements. The PSAR should describe those measures that assure that qualifications of special processes, personnel performing special processes, and equipment are kept current and that record files thereof are maintained.

17.1.10 Inspection

17.1.10.1. The PSAR should describe the measures that assure that a program for inspection is established and implemented by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The PSAR should describe measures that assure that (1) inspection personnel are appropriately qualified and are independent of the individual or group performing the activity being inspected, (2) inspections or tests are performed for each work operation as necessary to verify quality, (3) indirect control by monitoring processing methods, equipment, and personnel is used if direct inspection of processed material or products is impossible or disadvantageous, and (4) both inspection and process monitoring are used when control is inadequate without both. The PSAR should describe measures that assure that (1) inspection procedures and instructions are made available with necessary drawings and specifications for use prior to performing the inspections, (2) inspectors' qualifications or certifications are kept current, (3) replaced or reworked items are inspected in accordance with original inspection requirements, and (4) modified or repaired items are inspected by methods that are equivalent to the original inspection method.

17.1.10.2. The PSAR should describe the system whereby appropriate documents will identify any mandatory inspection holdpoints that require witnessing or inspecting by the applicant or his designated representative and beyond which work may not proceed without the consent of his designated representative.

17.1.11 Test Control

17.1.11.1. The PSAR should describe the measures that establish a test program that (1) identifies all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service, (2) is conducted by trained and appropriately qualified personnel in accordance with written test procedures that incorporate or reference the requirements and acceptance limits contained in applicable design documents, and (3) includes testing that will be performed under the construction permit.

17.1.11.2. The PSAR should describe the measures that assure test procedures have provisions for assuring that:

1. All prerequisites for the given test have been met,

2. Adequate test instrumentation and equipment are available, and

3. The test is performed under suitable environmental conditions and with adequate test methods.

17.1.11.3. The PSAR should describe the system whereby test results are documented and evaluated to assure that test requirements have been satisfied.

17.1.12 Control of Measuring and Test Equipment

The PSAR should describe the measures established to assure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly identified, controlled, adjusted, and calibrated at specified periods to maintain accuracy within necessary limits. The PSAR should describe measures that assure (1) that these devices are adjusted and calibrated against certified equipment or reference or transfer standards having known valid relationships to nationally recognized standards or (2) that if no national standards exist, the basis for calibration is documented. The PSAR should describe the measures that assure that the error of calibration standards is less than the error of production measuring and test equipment. The PSAR should describe provisions that will apply if measuring and test equipment is found out of calibration (1) for evaluating the validity of previous inspection or test results and the acceptability of items inspected or tested since the last calibration check and (2) for repeating original inspections or tests using calibrated equipment where necessary to establish acceptability of suspect items. The PSAR should describe measures that assure the maintenance of records that indicate the calibration status of all items under the calibration system and that identify the measuring and test equipment.

17.1.13 Handling, Storage, and Shipping

The PSAR should describe the measures established to control the handling, storage, shipping, cleaning, and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. The PSAR should describe the measures for specifying and providing, when necessary for particular products, special protective environments such as inert gas atmosphere, specific moisture content levels, and temperature levels.

17.1.14 Inspection, Test, and Operating Status

The PSAR should describe measures established to indicate by the use of markings such as stamps, tags, labels, routing cards, or other suitable means the status of inspections and tests performed on individual items of the nuclear power plant throughout fabrication, installation, and test. The PSAR should describe measures that provide for the identification of items that have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of such inspections and tests. The PSAR should describe the measures established for indicating the operating status of structures, systems, and components of the nuclear power plant such as tagging valves and switches to prevent inadvertent operation.

17.1.15 Nonconforming Materials, Parts, or Components

The PSAR should describe the measures established to control materials, parts, or components that do not conform to requirements in order to prevent their inadvertent use or installation. The PSAR should describe measures that provide for, as appropriate, identification, documentation, segregation, disposition, and notification to affected organizations. The PSAR should describe measures that assure that nonconforming items are reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures. The PSAR should describe measures that control further processing, delivery, or installation pending proper disposition of the deficiency. The PSAR should describe measures established by the applicant (1) for contractors to report to him those nonconformances concerning departures from procurement requirements that are dispositioned "use as is" or "repair" and (2) to make such nonconformance reports part of the documentation required at the nuclear plant site or to include a description of the nonconformance and its disposition on certificates of conformance that are provided to the site prior to installation or use of material or equipment at the site. The PSAR should state whether periodic analyses of nonconformance reports are performed to show quality trends and whether such analyses are forwarded to management.

17.1.16 Corrective Action

17.1.16.1. The PSAR should describe the measures that assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

17.1.16.2. The PSAR should describe how, in the case of significant conditions adverse to quality, the cause of the condition is determined, corrective action is taken to preclude repetition, and the problem with its determined cause and corrective action is documented and reported to appropriate levels of management.

17.1.17 Quality Assurance Records

17.1.17.1. The PSAR should describe the measures that assure that sufficient records are maintained to furnish evidence of activities affecting quality. The PSAR should describe how the content of such records (1) includes at least the following: test logs; results of reviews, drawings, inspections, tests, audits, monitoring of work performance, and materials analyses; and such data as qualifications of personnel, procedures, and equipment; (2) identifies the type of operation, the inspector or data recorder, the results, the acceptability, and action taken in connection with any deficiencies noted; and (3) provides sufficient information to permit identification of the record with the item or activity to which it applies.

17.1.17.2. The PSAR should describe the measures that assure that records will be identifiable and retrievable.

17.1.17.3. The PSAR should describe the measures that establish requirements (consistent with regulatory requirements and responsibilities concerning record submittal and retention, security, and storage facilities) for protecting records from destruction by fire, flooding, tornadoes, insects, and rodents and from deterioration by extremes in temperature and humidity.

17.1.18 Audits

The PSAR should describe the program of the applicant and of the principal contractors for conducting comprehensive planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program.

The PSAR should describe the program features that cover the functions listed below and should identify the positions or organizations that perform these functions.

1. External audits to be performed by the applicant and his principal contractors on their respective suppliers,

2. Internal audits to be performed by the applicant and his principal contractors within their respective organizations,

3. The planning and scheduling of audits to assure that they are regularly scheduled on the basis of the status and safety importance of the activities being performed and are initiated early enough to assure effective quality assurance during design, procurement, manufacturing, construction and installation, inspection, and testing,

4. Conduct of audits in accordance with written procedures or check-lists by appropriately trained and qualified personnel not having direct responsibility in the area being audited, and

5. Documentation of audit results with review by management responsible for the area audited and, where indicated, followup action taken, including re-audit of the deficient areas.

17.2 Quality Assurance (QA) During the Operations Phase

The FSAR should describe the QA program that will assure the quality of all safety-related items and activities during the operations phase. These activities include plant operation, maintenance, repair, inservice inspection, refueling, modifications, testing, and inspection under the operating license.

The description of the QA program in the FSAR should include the applicable information requirements outlined in Section 17.1 (i.e., substitute "FSAR" for "PSAR" in 17.1, above), except for those activities applicable only to the construction phase (activities performed under the construction permit). The FSAR should describe the QA program under the cognizance of the offsite and onsite QA organizations and should show that it addresses each of the criteria of Appendix B to 10 CFR Part 50. The description should delineate any significant differences in functional responsibilities between the offsite and onsite QA organizations.

The FSAR should describe the extent to which the operations phase QA program will follow the guidance in WASH-1284, "Guidance on QA Requirements During the Operations Phase of Nuclear Power Plants," and the extent to which activities involving design, procurement, and construction during the operations phase will follow the guidance in WASH-1283, "Guidance on QA Requirements During Design and Procurement Phase of Nuclear Power Plant," and in WASH-1309, "Guidance on QA Requirements During the Construction Phase of Nuclear Power Plants." If such guidance will not be followed, the applicant should describe acceptable alternative methods in detail equivalent to that furnished in the above guidance.

APPENDIX A*

INTERFACES FOR STANDARD DESIGNS

Table of Contents

*Although Appendix A is a new addition to Regulatory Guide 1.70, Revision 3, the lines indicating changes have not been included.

ABBREVIATIONS FOR NUCLEAR PLANT SYSTEMS

ADS	-	Automatic depressurization system
BOP	-	Balance of plant (all systems, structures, and components comprising a total plant excluding the NSSS and site- and utility-specific items)*
CCWS	-	Component cooling water system
CIS	-	Containment isolation system
CVCS	-	Chemical and volume control system
ECCS	-	Emergency core cooling systems
ESF	-	Engineered safety feature
GWMS	-	Gaseous waste management system
I&C	-	Instrumentation and control
LWMS	-	Liquid waste management system
MSIVLCS	-	Main steam isolation valve leakage control system
MSLIV	-	Main steam line isolation valve
NSSS	-	Nuclear steam supply system (components and piping comprising the RCS and directly related auxiliary systems)*
PRDS	-	Pressurizer relief discharge system
RCICS	-	Reactor core isolation cooling system
RCPB	-	Reactor coolant pressure boundary
RCS	-	Reactor coolant system
RHRS	-	Residual heat removal system
RWCUS	-	Reactor water cleanup system
SSWS	-	Station service water system
SWMS	-	Solid waste management system

*See Amendment 1 to WASH 1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants."

INTERFACES FOR STANDARD DESIGNS

I. INTRODUCTION

Safety-related interfaces must be identified and defined for standard designs submitted under Option 1 (Reference Systems) of the Commission's standardization policy to establish the requirements that must be met and assumptions that must be verified by other unspecified portions of a nuclear plant design to ensure that systems, components, and structures within the standard design will perform their safety functions. Safety-related interfaces also include information that may be useful in the design and staff review of the unspecified portions of the plant design. The safety functions of a standard design are those essential functions that ensure (1) the integrity of the reactor coolant pressure boundary; (2) that the specified acceptable fuel design limits are not exceeded as a result of anticipated transients; (3) the capability to shut down the reactor and maintain it in a safe shutdown condition; and (4) the capability to prevent or mitigate the consequences of an accident that could result in radiation exposures in excess of applicable guidelines. Interfaces are used, therefore, to provide a basis for ensuring that the matching portions of a nuclear plant design, as described in a PSAR for a CP application that references the standard design or in another Standard Safety Analysis Report (SSAR) for a matching portion of the plant, are compatible with the standard design regarding the safety-related aspects of the plant design.

This appendix describes safety-related interfaces, for light-water reactors only, that should be presented at the preliminary design* stage of review by the reactor vendor in a Nuclear Steam Supply System SSAR (NSSS-SSAR)** and by the architect-engineer in a Balance-of-Plant SSAR (BOP-SSAR).** The interfaces for a BOP-SSAR are also directly applicable to an SSAR describing an entire nuclear plant (NSSS plus BOP, but excluding utility- and site-specific items). This appendix also describes an acceptable format for presenting interfaces in an SSAR.

Criteria for determining the acceptability of interfaces, as necessary for safety, are not included in this appendix. While not identified specifically as interface acceptance criteria, the criteria are part of other guidance already made available by the NRC, including that contained in the regulations, regulatory guides, and codes and standards.

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*Many of the interfaces identified in this document are also applicable at the final design stage of review. Definitive guidance regarding all final design interfaces will be provided later as the need arises.

**The specific interface items presented herein apply only to an NSSS/BOP division of design scope for a nuclear plant or to an entire nuclear plant (NSSS and BOP); they do not apply to any other division of design scope such as a nuclear island/turbine island.

The compilation of interfaces presented in this appendix is based on staff consideration of all safety criteria applicable to the review of nuclear power plant designs, including those contained within the regulations, regulatory guides, and codes and standards, and on the background and experience acquired during the staff review of the several standard design applications already submitted. In this light, the staff considers the present listing of interfaces to be essentially complete and to promote maximum flexibility for design and for component selection consistent with the requirements for safety. However, standard design applications and utility applications referencing standard designs should not necessarily be limited to the interfaces listed; any additional interfaces determined to be important to safety should be identified and addressed in these applications, especially those interfaces that may be unique to a particular plant design. It is also the staff's intent to supplement and revise the interface lists, as well as other aspects of this appendix, as additions and modifications are indicated.

II. SOURCES OF INTERFACES

Interfaces for standard designs stem from the following sources:

a. Requirements for safe operation of the standard design that must be satisfied by matching portions of the plant design or by the utility (e.g., cooling water and electric power requirements for the NSSS that must be provided by the BOP, an inservice inspection program for the NSSS and BOP that must be provided by the utility).

b. Assumptions made for the standard design that must be more precisely defined during the design coordination effort between the reactor vendor and the architect-engineer or between the architect-engineer and the utility (e.g., mass and energy release rates during a LOCA specified by the reactor vendor that must be coordinated with the containment design provided by the architect-engineer).

c. Site-related design assumptions upon which the standard design is based.

d. Criteria pertinent to the standard design described in the SSAR under review that may be useful for the design and staff review of matching systems, components, and structures (i.e., safety criteria for the items within the standard design, including codes and standards, General Design Criteria, and regulatory guides).

Each of the above sources was used by the staff in preparing the lists of interfaces shown in Sections VI and VII and, in turn, should be used by reactor vendors and architect-engineers when identifying and defining interfaces for presentation in SSARs.

III. INTERFACES TO BE ADDRESSED IN SSARs

The interface items that should be addressed in SSARs for both normal and abnormal operating conditions have been identified by the staff as shown in Sections VI and VII. Those interfaces listed in Section VI should be defined by reactor vendors in NSSS-SSARs. Those listed in Section VII should be defined by architect-engineers in BOP-SSARs or by other organizations in standard plant SSARs (NSSS plus BOP). In addressing these interfaces, the standard design applicant should clearly define the scope of design encompassed by the SSAR; the definition so provided should be consistent with the gross definition of the content of NSSS-SSARs and BOP-SSARs as given in Amendment 1 to WASH-1341. The sources of the interfaces listed in Sections VI and VII are items a, b, c, and d described in Section II above.

IV. USE OF INTERFACES PRESENTED IN SSARs

All interfaces presented in an SSAR should be addressed in a referencing SAR (either a PSAR for a CP application or another SSAR) that describes the matching portion of the nuclear plant design. The description of the matching portion should clearly indicate that each interface has been recognized, used, and satisfied by the design of the interfacing system, component, or structure. For those interfaces involving a design coordination effort between the reactor vendor and the architect-engineer, the utility application should clearly describe the outcome of this effort in terms of the resulting design of the interfacing systems, components, and structures. In this way, the compatibility of matching portions of a plant design with regard to licensing requirements is demonstrated.

For site interfaces, the utility PSAR referencing a standard design should demonstrate that the site design parameters established as the basis for the standard design envelop the characteristics of the proposed plant site described in the CP application (e.g., the response spectrum used for the seismic design of the standard design applied at the foundation level should be shown to envelop the response spectrum derived at the foundation level for the proposed site).

It should be noted that acceptability to the staff of the compatibility of matching portions of a plant design with regard to licensing requirements in no way relieves the utility-applicant that has referenced a standard design in his application for licenses from his responsibility under the NRC regulations to ensure that all interfaces between matching systems, components, and structures are satisfied for compatibility.

V. FORMAT FOR INTERFACE PRESENTATION IN SSARs

Interfaces should be presented in an SSAR in a manner that will facilitate their location by staff reviewers and other groups involved in the licensing and design processes. In addition, interfaces should be presented on a system-by-system basis consistent with the approach for presenting plant design information established in Regulatory Guide 1.70. The following guidance for presenting interfaces in SSARs describes an acceptable format to accomplish these purposes:

a. Chapter 1, "Introduction and General Description of Plant," should include an interface section presenting an overall "road map" matrix to guide the reviewer to other sections in the SSAR where the specific interfaces can be found. The matrix should include:

1. A listing of all systems and structures within the standard design that interface with matching unspecified portions of the plant;

2. A listing of other interface areas that can be referenced in support of the items listed in 1 above;

3. A listing of the particular items in the matching unspecified portion with which the standard design items interface (e.g., the RHRS in the NSSS-SSAR interfaces with the CCWS, emergency onsite power system, containment sump, and refueling water storage tank in a BOP-SSAR or a PSAR); and

4. Identification of the section in the SSAR in which the specific interfaces are described.

Examples of an acceptable approach for preparing the matrices for an NSSS-SSAR and a BOP-SSAR are shown in Figures 1 and 2, respectively.

b. Specific interfaces should be presented on a system-by-system basis to the maximum extent practicable and should be shown in a separate subsection (as identified in Sections VI and VII and in the Table of Contents for Regulatory Guide 1.70) directly associated with the system description (not in the section assigned by Regulatory Guide 1.70 for the system description). The subsection should incorporate drawings, piping and instrumentation diagrams, and tables either directly or by reference (provided the interfaces intended to be referenced are clearly indicated therein). In general, descriptive material in other sections that may contain interfaces should not be referenced.

c. Interfaces of a broader nature that apply to classes of systems, components, or structures (e.g., items 3.4.3, 3.5.4, 3.6.3, 3.7.5, 3.8.6, 3.9.7, 3.10.5, and 3.11.6 in Section VI for an NSSS-SSAR and items 3.3.3, 3.4.3, 3.5.4, 3.7.5, 3.8.6, and 3.9.7 in Section VII for a BOP-SSAR) should be presented in the appropriate sections of other chapters of the SSAR (i.e., Chapter 3). These are supporting interfaces that should, in turn, be referenced in the interface subsections for the systems and structures

Figure 1

EXAMPLE OF MATRIX OF INTERFACE AREAS FOR AN NSSS-SSAR

Items on Matching Portion of Plant

NSSS Interface Areas	Feedwater System	Main Steam System	Component Cooling Water System	Offsite Power System	Onsite A.C. Power System	Auxiliary F.W. System	Containment	Site	Ventilation System	Liquid Waste Management System	Control Room	D.C. Power Supply	Location in SSAR
1. SYSTEM INTERFACE AREAS													
Reactor Coolant Pressure Boundary	X	X	X	X		X	X				X		5.2.6
Emergency Core Cooling Systems			X	X	X		X				X	X	6.3.6
Reactor Trip System		X		X	X		X				X	X	7.2.3
Habitability Systems			X								X		6.4.7
Fuel Handling System							X		X				9.1.4.6
Standby Liquid Control System				X	X						X	X	9.3.5.6
Chemical and Volume Control System			X	X					X	X	X		9.3.4.6
Reactor Water Cleanup System			X	X					X	X	X		5.4.8.4
Residual Heat Removal System			X	X	X		X		X		X	X	5.4.7.5
2. SUPPORTING INTERFACE AREAS													
Flood Protection													3.4.3
Missile Protection													3.5.4
Pipe Whip Protection													3.6.3
Mechanical Systems and Components													3.9.7
Environmental Design of Mechanical and Electrical Equipment													3.11.6
Inservice Inspection of Class 2 & 3 Components													6.6.9
Fire Protection													9.5.1.6
Safety Actions by BOP													15.X.X

Figure 2

EXAMPLE OF MATRIX OF INTERFACE AREAS FOR A BOP-SSAR

Site- and Utility-Specific Items

BOP Interface Areas	Switchyard	Ultimate Heat Sink	Intake Structure	Inservice Insp. Program	Initial Test Program	Industrial Security	10 CFR Part 50 App. I	Meteorology	Seismic Design Parameter	Wind & Tornado Parameter	Geology	Probable Maximum Flood	Location in SSAR
1. INTERFACE AREAS FOR SYSTEMS, COMPONENTS, AND STRUCTURES													
Station Service Water System		X	X	X	X	X			X			X	9.2.1
Instrumentation and Control			X		X				X				7.8
Fire Protection Program	X	X	X		X				X				9.5.1.6
Onsite A.C. Power System	X		X		X	X			X				8.3.1.5
Water Systems		X	X	X	X								9.2.7
Liquid Waste Management System		X		X	X		X						11.2.4
Gaseous Waste Management System				X	X		X	X					11.3.4
Effluent Monitoring and Sampling								X					11.5.3.X
Other Auxiliary Systems		X		X	X								9.5.9
2. SUPPORTING INTERFACE AREAS													
Wind and Tornado Loadings										X			3.3.3
Water Level Design												X	3.4.3
Seismic Design									X				3.7.5
Design of Category I Structures								X	X	X	X	X	3.8.6
Industrial Security						X							13.6.3
Mechanical Systems and Components				X	X								3.9.7

described in the SSAR. Examples include site design parameters, protection against missiles, and protection against pipe whip.

d. For an NSSS-SSAR, the interfaces identified in Section VI should be presented for each system (review area) in the following categories to facilitate review. Review areas in Section VI of a broad nature, as discussed in item c above, need not use these categories for presentation of interfaces.

1. Power - Requirements for all types of power for safety systems and components.

2. Protection Against Natural Phenomena - Requirements for protection of safety-related systems and components against naturally occurring events such as earthquakes, wind, tornadoes, and floods.

3. Protection Against Effects of Pipe Failure - Requirements for protection of safety-related systems, components, and structures inside and outside containment against the dynamic effects resulting from the failure of piping in high- and moderate-energy systems, including pipe whip, jet impingement, and other dynamic effects.

4. Missiles - Requirements for protection of reactor coolant pressure boundary and other safety-related systems against internally generated missiles and missiles generated by naturally occurring events both inside and outside containment; identification of potential missiles from NSSS equipment.

5. Separation - Requirements for physical separation to prevent a single event from causing failure of redundant safety systems and components.

6. Independence - Requirements for independence to prevent a failure in a safety system or component from causing failure in its redundant safety system or component.

7. Thermal Limitations - Requirements for heating or cooling of safety systems and components, including fluid conditions and limitations.

8. Monitoring - Requirements for performance surveillance, testing, and inspection of safety systems and components (technical specifications not included).

9. Actuation/Controls - Requirements for actuation of safety systems and components, for control of their subsequent operation, and for interlocks.

10. Chemistry/Sampling - Requirements for fluid chemistry, purity, and sampling for safety systems and components.

11. Materials - Requirements for materials for safety systems and components.

12. System/Component Arrangement - Location requirements (including inservice inspection and testing) that safety systems and components place on plant arrangement.

13. Radioactive Waste - Source term characteristics for the collection, treatment, and disposal of radioactive wastes.

14. Related Service - Requirements for other essential services for safety systems and components (e.g., interfaces from Section V.f below, fire protection, compressed air).

15. Overpressure Protection - Requirements for ensuring that pressure limits for safety systems are not exceeded.

16. Environment - Requirements for environmental conditions that must be provided for proper operation of safety systems and components.

17. Mechanical Interaction Between Systems - Requirements for consideration of differential motion, including seismic effects and thermal expansion.

18. Design Criteria - Criteria upon which the NSSS system designs, or portions thereof, are based.

e. For a BOP-SSAR, interfaces identified in Section VII should be presented for each system (review area) in the following categories to facilitate review. Review areas in Section VII of a broad nature, as discussed in item c above, need not use these categories for presentation of interfaces.

1. Power - Requirements for all types of power for safety systems and components (e.g., offsite power to plant during certain conditions, power from plant to site-specific components).

2. Site Parameters - Site design parameters, based on site characteristics (seismic, geological, hydrological, and meteorological), used for the design of safety systems, components, and structures against naturally occurring events.

3. Missiles - Missiles generated by natural phenomena used as the basis for the design of safety systems, components, and structures.

4. Thermal Limitations - Requirements for cooling safety systems and components, including fluid conditions and limitations.

5. Monitoring - Requirements for performance surveillance, testing, and inspection of interfacing safety systems and components.

6. Actuation/Controls - Requirements for actuation of interfacing systems and components and for control of their subsequent operation.

7. Materials - Requirements for materials for safety systems and components.

8. System/Component Arrangement - Location requirements (including inservice inspection and testing) that safety systems and components place on plant arrangement.

9. Radioactive Waste - Release of radioactive material to the environment.

10. Related Service - Requirements for other essential services for safety systems and components (e.g., interfaces from Section V.f below, fire protection, compressed air).

11. Mechanical Interaction Between Systems and Buildings - Requirements for consideration of differential motion, including seismic effects and thermal expansion.

12. Design Criteria - Criteria upon which BOP system designs, or portions thereof, are based.

f. The physical points of interface for fluid systems should be indicated on piping and instrumentation diagrams (P&IDs) and those for electric systems on elementary, schematic, or logic diagrams and on block diagrams to clearly show the line of demarcation between the standard design and the unspecified matching portions of the plant. Each system interfacing point should be uniquely labeled. The selection of specific interface points should be based on the division of <u>design</u> responsibility, not <u>supply</u> responsibility, established between the reactor vendor and architect-engineer (e.g., an interface point should not be established between a component supplied by the reactor vendor and the piping in the same system supplied by the architect-engineer). Safety-related fluid and electric interfaces applicable to each point, as identified in Sections VI and VII, should be listed, consistent with the system-by-system basis established for the definition of interfaces (e.g., the compressed air requirements for all required air-operated valves in the system as a group, the d.c. power requirements for all d.c. instrumentation in the system as a group).

g. All the standard design interfaces should be addressed in a referencing SAR (either a PSAR for a CP application or another SSAR) for the design of an interfacing system, component, or structure; for the design of utility-specific items; or for the determination of standard plant/site compatibility. The specific interfaces used for each interfacing area should be identified. The identification should be presented in the "Design Bases" section of each system description. Identification should consist of appropriate references to the interfaces in the referenced SSAR; the specific interfaces should not be rewritten or reprinted. For interfaces to P&IDs and to electric diagrams in a referencing SAR, a similar procedure should be used.

VI. NUCLEAR STEAM SUPPLY SYSTEM INTERFACES

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.4 Water Level (Flood) Design

3.4.3 NSSS Interface

Safety-related NSSS equipment located outside containment that must be protected from flooding.

3.5 Missile Protection

3.5.4 NSSS Interface

1. NSSS equipment located inside or outside containment that potentially could produce missiles, including type (e.g., valve bonnet, studs, stems, thermowells), weight, size, and energy of each missile.

2. Safety-related NSSS equipment located outside containment requiring protection from externally generated missiles (e.g., tornado missiles).

3. Safety-related NSSS equipment located inside or outside containment requiring protection against internally generated missiles.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.3 NSSS Interface

1. Identification of high- and moderate-energy NSSS pipelines inside and outside containment.

2. Safety-related NSSS systems and equipment located inside or outside containment requiring protection from the effects of failures of high- and moderate-energy pipelines.

3. The coordination of the design of the RCS with interfacing BOP-designed piping systems regarding postulated pipe break locations, orientation, configurations, and resulting loads to ensure compatibility.[1]

3.7 Seismic Design

3.7.5 NSSS Interface

1. A listing of NSSS systems and components that, in conjunction with supporting structures, are designed to Seismic Category I requirements. (Information given in Section 3.2.1 of the SAR may be referenced here.)

2. The establishment at all support points of the seismic response spectra envelopes to which the standard NSSS equipment is designed for use in the BOP design.[1]

3. Envelopes of allowable seismic loads transmitted from Category I or non-Category I systems that connect to the standard NSSS components for use in the BOP design.[1]

4. The mass and stiffness properties of the NSSS to be coupled with the mathematical model of the seismic system, including structures and supports, for use in the BOP design.[1]

3.8 Design of Category I Structures

3.8.6 NSSS Interface

1. The maximum differential displacements and rotations due to the loads at points of the NSSS that will interface with BOP structures for use in the BOP design.[1]

2. The range of structural properties of supporting BOP structures that were used in the analysis of the NSSS for use in the BOP design.[1]

3. All the loads that have to be transmitted from the NSSS components to the supporting BOP structures for use in the BOP design.[1]

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[1]This interface involves the exchange of information among the utility, the NSSS designer, and the BOP designer to ensure compatibility of interfacing systems, components, and structures. This information exchange takes place in accordance with the requirements of Appendix B to 10 CFR Part 50. The information need not be provided to the NRC unless specifically requested by the staff; however, the fact that such information was exchanged and the necessary evaluations were performed should be documented in the SSAR.

4. Evaluation of the deflections under all loading conditions, provided by the BOP designer for the BOP structures supporting NSSS components.

5. Evaluation of design information and drawings prepared by the BOP designer, as they affect NSSS systems and components to ensure compliance with NSSS design criteria.*

3.9 Mechanical Systems and Components

3.9.7 NSSS Interface

1. The coordination of the design of the RCS and interfacing BOP-designed systems, components, and supports when inelastic analysis methods are used by either the NSSS or BOP designer to ensure compatibility. Areas requiring coordination should include analytical criteria, procedures, and results.*

2. Preoperational piping vibration test parameters for the NSSS system and components for all ASME Class 1, 2, and 3 piping systems for use in the BOP design.*

3. The establishment of the test program for the flow-induced vibration of reactor internals for use by the utility. (Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," is applicable reference.)[2]

4. The coordination of the design of NSSS active and inactive components and their supports with the design of interfacing BOP components and supports regarding design loading combinations to ensure structural and functional compatibility. The categorization of the appropriate plant and component operating conditions should be coordinated with the BOP designer.*

5. The coordination of the structural and functional aspects of overpressure protection for NSSS-designed systems and components with the BOP designer to ensure compatibility.*

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[2]The staff recognizes that the information may not be available at the PDA stage of review. If all the information needed by the staff and ACRS to complete their review of this interface is not provided in the application for a PDA, the PDA will be subject to a condition that either the additional information be provided to the utility for inclusion in a CP application referencing the PDA or the utility must demonstrate in its CP application that such information may reasonably be left for later consideration in accordance with §50.35(a) of 10 CFR. Issuance of a PDA does not foreclose staff and ACRS review of interfaces subject to such a condition.

*See footnote 1 on page A-11.

6. Limiting criteria affecting NSSS active component operability for use in the BOP design.*

7. The development of reference test data for inservice testing of NSSS pumps and valves as specified in subsections IWP and IWV of ASME Section XI for use by the utility.*

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

3.10.5 NSSS Interface

The coordination of the seismic design requirements of all NSSS safety-related instrumentation and electrical equipment and supports with regard to the floor response spectra defined by the BOP designer to ensure compatibility.*

3.11 Environmental Design of Mechanical and Electrical Equipment

3.11.6 NSSS Interface

1. Heat loads and environmental requirements for NSSS equipment located outside containment.

2. Maximum and minimum containment environmental conditions (i.e., temperature, pressure, humidity, radiation level, etc.) to which NSSS safety-related mechanical and electrical equipment is qualified.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the design of the RCPB.

2. Quantity of reactor coolant transferred to secondary side of the steam generator following a tube failure; time to effect pressure equalization between a defective steam generator and the RCS; and minimum water volume and maximum steam volume on the secondary side of a steam generator during normal operation.

3. Steam and feedwater conditions (i.e., flow, pressure, and temperature) under all modes of operation, including startup and shutdown.

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*See footnote 1 on page A-11.

4. Minimum total capacity and maximum set pressures for secondary safety valves (ASME Code Section III), maximum accumulation, division of relief capacity among main steam lines (including maximum flow per valve at set pressure), and minimum or limiting steamflow for atmospheric relief valves for each main steam line, including pressure and temperature.

5. For PWRs only, design requirements for the piping connecting the pressurizer to the pressurizer relief tank (including maximum steam-flow to be accommodated and maximum back pressure at valve discharge).

6. Volume of reactor coolant contained within the RCPB.

7. Requirements for leak detection systems (e.g., type of leakage, locations, rates) to permit control room monitoring of identified and unidentified leakage from the RCPB to containment and of intersystem leakage from the RCPB (including leakage to the secondary side of a steam generator).

8. For BWRs only, mass and energy release rate data for safety and relief valve discharges during anticipated transients.

9. For BWRs only, assumed impulse loads to which NSSS system and components may be subjected due to pool swell forces during a LOCA blowdown. Coordination of these loads with the specific containment design provided by the BOP designer to ensure compatibility.

10. For PWRs only, maximum steam generator mass and energy release rate data for a spectrum of steam and feedwater line break sizes inside containment and selected plant operating conditions; and requirements for isolating flow to any secondary system pipe break, including MSLIV's closure time.

11. For PWRs only, mass and energy release rate data for selected RCS break sizes and locations and the assumed maximum containment design pressure used to generate the mass and energy release rate data. Coordination of the assumed pressure with the containment analysis performed by the BOP designer to ensure that the maximum containment pressure calculated by the BOP designer does not exceed the pressure assumed by the reactor vendor.*

12. For BWRs only, mass and energy release rate data for a spectrum of assumed main steam line and recirculation line break sizes of selected RCS piping locations; mass and energy release rate data for RHRS head spray line and RWCUS line breaks.

13. For PWRs only, requirements for auxiliary feedwater provisions to all intact steam generators, with or without offsite and normal onsite power available, assuming isolation of a steam generator, if applicable, due to a pipe break event such as a steam line break, feedwater line break, or a steam generator tube rupture:

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*See footnote 2 on page A-12.

a. Reliability requirements (i.e., redundancy, diversity, etc.),

b. Minimum flow capability and maximum flow to a single steam generator,

c. Minimum discharge pressure,

d. Maximum time to attain full flow following demand signal,

e. Minimum volume of stored condensate or standby auxiliary feedwater required to bring reactor to cold shutdown with zero time at hot standby and no allowance to maintain reactor at cold shutdown,

f. Temperature limits,

g. Quantity and condition of steam available from intact steam generator for motive power (final values to be coordinated with the BOP designer to ensure compatibility),

h. Conditions within the NSSS that initiate flow,

i. Conditions within the BOP that are assumed to initiate flow, and

j. Requirements for compatibility of the control and power systems with the NSSS actuation and redundancy logic.

14. Maximum stroke time for MSLIV; requirements for the location, capacity, and control arrangement of the main steam line relief and dump valves and other main steam system valves located between the MSLIV and the main condenser; and requirements for the feedwater control system.

15. Limiting heat loads and coolant conditions (flow, pressure, and temperature) for the condensate storage facilities for all plant modes of operation, including accident conditions; and minimum water inventory for cold shutdown.

16. For PWRs only, temperature, pressure, radioactivity concentrations, and flow rate to the steam generator blowdown system during normal and anticipated operational occurrences; and isolation requirements.

17. For PWRs only, requirements to maintain secondary side water chemistry for steam generators within specified ranges (including steam generator blowdown, chemical addition, condensate purification, and monitoring).

18. Requirements to provide capability for sampling to monitor fluid system performance (including instrumentation for monitoring impurity

removal and for detecting excessive chloride and fluoride content). For PWRs only, requirements to sample and analyze reactor coolant for specified parameters at hot leg, pressurizer surge line, and pressurizer steam space. (Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel," for PWRs and 1.56, "Maintenance of Water Purity in Boiling Water Reactors," for BWRs are applicable references.)

19. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

20. Reactor coolant radioactivity concentrations, noble gas release rates (BWRs only), and leak rates from RCS to floor drains and building atmospheres.

21. Flow rate, batch volume, radioactivity concentrations, temperature, pressure, and partition factors at each RCS interface point with the process sampling system and for each leakage point to the building atmosphere during normal and anticipated operational occurrences.

22. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each RCS component interfacing with the SSWS or the CCWS.

23. Materials Interfaces 1 to 6 and 8, Table 1.

24. Locations and accessibility requirements for inservice inspection of ASME Code Class 1 components within RCPB. (ASME Code Section XI is applicable reference.)

25. Locations and accessibility requirements for inservice inspection of reactor coolant pump flywheels. (Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," is applicable reference.)

26. Locations and accessibility requirements for inservice inspection of steam generator tubes. (Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," is applicable reference.)

27. Criteria for contamination protection and cleaning before, during, and after welding installation of steam generators at NSSS-BOP boundaries (to avoid stress-corrosion cracking of Inconel tubes). (Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," is applicable reference.)

28. To preclude adverse effects on NSSS equipment, compatibility requirements for materials to be used in containment spray system, considering reactor coolant and radiation environment during accident conditions. (Austenitic stainless steel not sensitized or alternative steel specified

Table 1

MATERIALS INTERFACES COMMON TO SEVERAL SYSTEMS

1. Criteria for contamination protection and cleaning before, during, and after field welding installation of austenitic stainless steel components at NSSS-BOP boundaries.* (Regulatory Guides 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and 1.44, "Control of the Use of Sensitized Stainless Steel," are applicable references.)

2. Requirements for control of sensitization of field installation welds joining austenitic stainless steel components at NSSS-BOP boundaries. (Regulatory Guide 1.44 is applicable reference.)

3. Requirements for control of delta ferrite in field installation welds joining austenitic stainless steel components at NSSS-BOP boundaries. (Regulatory Guides 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and 1.44 are applicable references.)

4. Welding requirements for field welding installation of ferritic steel components and austenitic stainless steel components at NSSS-BOP boundaries, including preheat temperature control, welding materials, and clad welding requirements. (Regulatory Guides 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," and 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," are applicable references.)

5. Requirements for low halide nonmetallic thermal insulation on austenitic stainless steel at NSSS-BOP boundaries. (Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," is applicable reference.)

6. Material requirements for BOP piping connected to all fluid systems of the NSSS, including RCS. (Regulatory Guide 1.44 is applicable reference.)

7. Requirements to provide capability for sampling to monitor fluid system performance.

8. Requirements to provide capability for fluid purity and chemistry control within specified ranges during operation. (Regulatory Guides 1.44 for PWRs and 1.56, "Maintenance of Water Purity in Boiling Water Reactors," for BWRs are applicable references.)

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*NSSS-BOP boundary is a boundary between an NSSS system and a BOP system or a boundary between an NSSS component and a BOP component within an NSSS system.

by NSSS designer for metals contacting reactor coolant and materials resistant to environment.) (Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," is applicable reference.)

29. Heat loads and environmental limitations for NSSS systems and components.

30. Reactor coolant pump trip signals initiated by main turbine stop valve closure.

5.4 Component and Subsystem Design

5.4.6 Reactor Core Isolation Cooling System

5.4.6.5 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RCICS included within the NSSS.

2. Net positive suction head requirements at RCICS pump suction and required heat removal capacity (including tube-side coolant conditions) during all conditions of standby and shutdown cooling until reactor vessel is depressurized.

3. Mass and energy release rates for RCICS line breaks.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each RCICS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each RCICS component interfacing with the SSWS.

6. Materials Interfaces 1 to 8, Table 1.

7. Requirements for safety and relief valve back pressure, spatial separation, and discharge piping.

5.4.7 Residual Heat Removal System

5.4.7.5 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RHRS included within the NSSS.

2. Net positive suction head requirements at the RHRS pump suction and required heat removal capacity (including tube-side coolant conditions) during all conditions of shutdown cooling.

3. For BWRs only, mass and energy release rates for RHRS head spray line breaks.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each RHRS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each RHRS component interfacing with the CCWS or SSWS.

6. Materials Interfaces 1 to 8, Table 1.

7. Requirements for safety and relief valve back pressure, spatial separation, and discharge piping.

5.4.8 Reactor Water Cleanup System (BWRs)

5.4.8.4 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RWCUS included within the NSSS.

2. Mass and energy release rates for RWCUS line breaks.

3. Flow rate, batch volume, radioactivity content, and batch frequency for filters, filter sludges, demineralizer resins, and evaporator bottoms transferred from RWCUS equipment to the SWMS during normal and anticipated operational occurrences.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each RWCUS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Heat loads and service water flow, pressure, and temperature for normal and limiting conditions for each RWCUS component interfacing with the SSWS.

6. Materials Interfaces 1 to 8, Table 1.

5.4.11 Pressurizer Relief Discharge System (PWR)

5.4.11.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the PRDS included within the NSSS.

2. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each PRDS component interfacing with the CCWS.

3. Materials Interfaces 1 to 8, Table 1.

6. ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.4 Containment Isolation System

6.2.4.5 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the CIS included within the NSSS.

2. Definition of signals generated by NSSS equipment (i.e., safety injection, low vessel level, etc.) for use in developing diverse containment isolation signals, and characteristics of valves included in the NSSS design that are part of the CIS. Definition of BOP signals that indicate abnormal containment conditions and that must be coordinated with the NSSS in the design of the reactor shutdown systems to ensure compatibility.

3. Maximum leakage rate and type of fluid for all containment isolation devices included in the NSSS design.

4. Test fluid type and maximum quantity of test fluid required for testing of containment isolation devices included in the NSSS design.

5. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

6.3 Emergency Core Cooling System

6.3.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the ECCS included within the NSSS.

2. Maximum head loss (friction and elevation), minimum net positive suction head requirements for all ECCS pumps for all conditions of operation (including single failure, operator error, minimum containment ambient pressure, and long- and short-term cooling), and identification of all conditions under which ECCS must provide core cooling (e.g., single failure, flooding); consider recirculation mode using containment sump or suppression pool.

3. For PWRs only, design requirements for the piping connecting the accumulators to the RCS and provisions for nitrogen supply.

4. For BWRs only, design requirements for the piping connecting the ADS accumulator to the relief valves and provisions for air supply.

5. Design requirements for manually operated valves in the ECCS; requirements for straight piping runs for flow measuring devices in ECCS; limitations on total water volume in the RCS cold leg up to the ECCS check valves; maximum time to achieve full ECCS flow in the event of a LOCA (with and without the availability of normal a.c. power supply); limitations on particle size of impurities in ECCS water; requirements for venting and filling provisions for air removal to preclude water hammer events; and design capability for preoperational testing to demonstrate all aspects of system operability.

6. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

7. Hydrogen released by zirconium water reaction in core; maximum amount of hydrogen dissolved in RCS water during plant operations (PWRs only); hydrogen generated by radiolysis of water in the reactor and in the containment sump as a function of time after LOCA; surface area, weight, and thickness of aluminum and zinc provided as a part of NSSS equipment inside containment; and hydrogen generation rate due to corrosion of aluminum and zinc by containment spray post-LOCA. (Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," is applicable reference.)

8. For PWRs only, assumed containment parameters are:

   a. Maximum passive heat sinks (materials of construction, surface area, thickness),

   b. Maximum free volume in containment,

   c. Containment initial conditions (temperature, pressure, and humidity), and

   d. Maximum containment active heat removal capability (heat removal rates, start times, containment spray flow rate and temperature, etc.).

Minimum containment pressure analysis must be coordinated with the BOP designer using actual containment parameters to ensure compatibility.

9. For BWRs only, minimum containment pressure assumed is 14.7 psia; actual pressure must be determined in coordination with the BOP designer using actual containment parameters to ensure compatibility.

10. Requirements on recirculation water pH for emergency core cooling and containment cooling.

11. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each ECCS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

12. Heat loads, cooling water flow, pressure, and temperature for normal and limiting conditions for each ECCS component interfacing with the SSWS or the CCWS.

13. Materials Interfaces 1 to 8, Table 1.

14. Requirements for safety and relief valve back pressure, spatial separation, and discharge piping.

6.4 Habitability Systems

6.4.7 NSSS Interface

1. Safety-related NSSS control equipment located in the control room.

2. Limiting design and operational requirements of NSSS control equipment (e.g., temperature, humidity).

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.9 NSSS Interface

Locations and accessibility requirements for inservice inspection of all ASME Code Class 2 and 3 components within the NSSS auxiliary systems and ESFs. (ASME Code Section XI is applicable reference.)

6.7 Main Steam Line Isolation Valve Leakage Control System (BWRs)

6.7.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the MSIVLCS included within the NSSS.

2. MSIVLCS parameters, including MSLIV leak rate, concentrations of radioactivity in steam, and setpoints on instrumentation, to be interlocked with leakage control system. (Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," is applicable reference.)

3. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each MSIVLCS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Materials Interfaces 1 to 8, Table 1.

7. INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.3 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RTS included within the NSSS.

2. Requirements for anticipatory trips (e.g., turbine trip signals as input to the reactor trip system).

7.8 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the I&C system included within the NSSS.

2. For each NSSS system, requirements for NSSS instrumentation energized by the plant instrumentation power supply system:

   a. Steady-state load,

   b. Transient/step load,

   c. Nominal system voltage,

   d. Allowable voltage regulation,

   e. Allowable harmonic content,

   f. Allowable frequency fluctuation,

g. Grounding requirements,

h. Power supply assignment, and

i. Percent peak deviation from true sine wave (inverters).

3. For each NSSS system, requirements for BOP sensors that provide inputs to accomplish NSSS functions and for associated instrument lines:

a. Range (including accident conditions),

b. Measurement accuracy,

c. Repeatable accuracy,

d. Maximum expected transient,

e. Response time (maximum allowable time to achieve sensor output after reaching trip level for measured variable),

f. Trip setpoint,

g. Snubbers,

h. Orifice,

i. Arrangement for instrument lines,

j. Type and location of readout, and

k. Bypass and inoperable status indication.

4. Number of logic trains used for the control of safety systems.

8. ELECTRIC POWER

8.2 Offsite Power System

8.2.3 NSSS Interface

For each NSSS system, requirements for offsite a.c. power:

1. Steady-state load,
2. Inrush kVA for motor loads,
3. Nominal voltage,
4. Allowable voltage regulation,
5. Nominal frequency,

6. Allowable frequency fluctuation,
7. Maximum frequency decay rate and limiting underfrequency value for reactor coolant pump coastdown, and
8. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

8.3 Onsite Power Systems

8.3.1 A.C. Power Systems

8.3.1.5 NSSS Interface. For each NSSS system, requirements for onsite a.c. power:

1. Steady-state load,
2. Inrush kVA for motor loads,
3. Nominal voltage,
4. Allowable voltage drop (to achieve full functional capability within required time period),
5. Sequence and time to achieve full functional capability for each load,
6. Nominal frequency,
7. Allowable frequency fluctuation,
8. Number of trains, and
9. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

8.3.2 D.C. Power Systems

8.3.2.3 NSSS Interface. For each NSSS system, requirements for onsite d.c. power:

1. Steady-state load,
2. Surge loads (including emergency conditions),
3. Load sequence,
4. Nominal voltage,
5. Allowable voltage drop (to achieve full functional capability within required time period),

6. Number of trains, and

7. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.4 NSSS Interface. Rack dimensions, weight, materials of construction, uplift forces, and mounting requirements; minimum storage capacity; minimum rack spacing and associated k_{eff} (with flooded unborated water and with optimum moderator aqueous foam); and vault drainage requirements.

9.1.2 Spent Fuel Storage

9.1.2.4 NSSS Interface

1. Rack dimensions, weight, materials of construction, uplift forces, and mounting requirements; minimum storage capacity; minimum rack spacing and associated k_{eff} (borated and unborated water); and allowable fuel pool water chemistry (e.g., pH, conductivity, boron concentrations) and limiting water temperature.

2. Minimum depth of water above spent fuel array to meet shielding requirements of 10 CFR Part 20; minimum depth of water above spent fuel bundle if accidentally dropped and positioned horizontally across top of spent fuel array; spent fuel pool normal and maximum decay heat loads (including fraction of core and minimum cooldown time prior to placing in the pool); and potential corrosion rate of racks and cladding and expected fission product leakage as a function of temperature and water chemistry.

9.1.4 Fuel Handling System*

9.1.4.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the Fuel Handling System included within the NSSS.

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*Fuel Handling System assumed to consist of the following NSSS items: bridge cranes; new fuel elevator; transfer tube and carriage; upender; and lifting rigs, slings, and other essential equipment integral to the NSSS.

2. Installation requirements (i.e., crane weights, power, compressed air, hydraulic requirements, etc.); storage requirements; capacity of bridge cranes and new fuel elevator; installation requirements for transfer tube, carriage, and upender; equipment interlocks and special built-in safety features; and other special requirements to preclude unacceptable accidents.

9.3 Process Auxiliaries

9.3.4 Chemical and Volume Control System (PWRs)

9.3.4.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the CVCS included within the NSSS.

2. Location of the CVCS letdown line radiation monitor and requirements to perform its alarm and control function.

3. Flow rate, batch volume, radioactivity content, and batch frequency for filters, filter sludges, demineralizer resins, and evaporator bottoms transferred from the CVCS equipment to the SWMS during normal and anticipated operational occurrences.

4. Flow rate, batch volume, radioactivity concentrations, temperature, pressure, and partition factors at each CVCS interface point with the GWMS and for each leakage point to the building atmosphere during normal and anticipated operational occurrences.

5. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each CVCS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

6. Heat loads and cooling waterflow, pressure, and temperature for normal and limiting conditions for each safety-related CVCS component interfacing with the CCWS.

7. Flow rate, boron concentrations, temperature, and pressure at each CVCS interface point with the refueling water or borated water storage tank and volume available in excess of requirements for accident condition.

8. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

9. Materials Interfaces 1 to 8, Table 1.

9.3.5 Standby Liquid Control System (BWRs)

9.3.5.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the SLCS included within the NSSS.

2. Boron concentration, flow rate, and requirements for maintaining minimum temperature.

3. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

4. Materials Interfaces 1 to 8, Table 1.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

9.5.1.6 NSSS Interface. Identification, quantification, and tabulation of the NSSS items that constitute a significant fire hazard.

10. STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features of Steam and Power Conversion System

10.4.4 Turbine Bypass System

10.4.4.X NSSS Interface

1. Steam conditions during discharge following turbine trip and limiting steamflow for turbine bypass system sizing without reactor trip, including temperature and pressure.

2. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

13. CONDUCT OF OPERATIONS

13.6 Industrial Security

13.6.3 NSSS Interface*

Identification of vital equipment as defined in 10 CFR §73.2 for use in developing physical security plans. Also, where applicable, locations

*This information should be submitted as proprietary information [10 CFR §2.790(d)].

of items of vital equipment and provisions incorporated into the design for monitoring the status of vital equipment to detect malevolent acts to impair performance.

14. INITIAL TEST PROGRAM

14.1 Specific Information To Be Included in Preliminary Safety Analysis Reports

14.1.8 NSSS Interface

Identification of the special or unique features of the NSSS design for consideration in developing the initial test program by the utility.

15. ACCIDENT ANALYSES

15.X Evaluation of Individual Initiating Events

15.X.X Event Evaluation

15.X.X.X NSSS Interface. Identification of the Safety Actions required to mitigate the consequences of each transient and accident event and the BOP system necessary to provide each Safety Action.

VII. BALANCE-OF-PLANT INTERFACES

2. SITE CHARACTERISTICS

2.3 Meteorology

2.3.6 BOP Interface

Limiting meteorological parameters (χ/Q) for design basis accidents (including use of backup hydrogen purge system) and for routine releases and other extreme meteorological conditions (e.g., temperatures, winds, humidity, dust storms, air quality, and appropriate combinations) for the design of systems and components exposed to the environment.

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.3 Wind and Tornado Loadings

3.3.3 BOP Interface

Tornado and operating basis wind loadings to which plant structures and exposed systems and components are designed.

3.4 Water Level (Flood) Design

3.4.3 BOP Interface

1. Flood and ground water elevation and requirements for protection established for the design of systems, components, and structures.

2. Hydrostatic loads established for the design of systems, components, and structures.

3.5 Missile Protection

3.5.4 BOP Interface

1. External missiles generated by natural phenomena established for the design of systems, components, and structures.

2. External missiles resulting from man-made hazards and accidents.

3.7 Seismic Design

3.7.5 BOP Interface

Seismic parameters ("g" values and response spectra) established for the design of systems, components, and structures. In addition, the range of site parameters (e.g., shear wave velocity, depth of embedment, depth of overburden) that are used in the seismic analysis should be specified.

3.8 Design of Category I Structures

3.8.6 BOP Interface

1. Snow, ice, and rain loads established for the design of systems, components, and structures.

2. Required bearing capacity of foundation materials, allowable absolute and differential settlements, and allowable tilt established for the design of systems, components, and structures.

3. Lateral earth pressure loads established for the design of systems, components, and structures.

4. Maximum loads, including overpressurization resulting from man-made hazards and accidents such as potential explosions and associated missiles in vicinity of the plant, potential aircraft impacts, etc.

3.9 Mechanical Systems and Components

3.9.7 BOP Interface

1. The preoperational piping vibration test parameters for use by the utility in developing the test programs for all ASME Class 1, 2, and 3 piping systems in the NSSS and BOP, including all high-energy piping systems outside containment and all Seismic Category I portions of moderate-energy piping systems outside containment.

2. The locations and other requirements for use by the utility in developing an inservice inspection program for ASME Class 1, 2, and 3 systems and components in the NSSS and BOP and for reactor coolant pump flywheels (PWRs only).*

3. Reference test data for inservice testing of BOP pumps and valves as specified in subsections IWP and IWV of ASME Section XI for use by the utility.

6. ENGINEERED SAFETY FEATURES

6.4 Habitability Systems

6.4.7 BOP Interface

Environmental conditions assumed in the design of the control room for protection of operators.

7. INSTRUMENTATION AND CONTROLS

7.8 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the I&C systems included within the BOP.

2. Provisions included in the plant instrumentation power supply system to accommodate the I&C requirements of the SSWS (and additional cooling capacity, if any required):

   a. Steady-state load,

   b. Transient/step load,

   c. Nominal system voltage,

   d. Allowable voltage regulation,

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*See footnote 2 on page A-12.

e. Allowable harmonic content,

f. Allowable frequency fluctuation,

g. Grounding requirements, and

h. Power supply assignment.

3. Provisions included for the sensors and their instrument lines associated with the SSWS (and additional cooling capacity, if any required) that provide inputs to satisfy station safety functions:

a. Range (including accident conditions),

b. Measurement accuracy,

c. Repeatable accuracy,

d. Maximum expected transient,

e. Response time (maximum allowable time to achieve sensor output after reaching trip level for measured variable),

f. Trip setpoint,

g. Snubbers,

h. Orifice, and

i. Arrangement for instrument lines.

8. ELECTRIC POWER

8.2 Offsite Power System

8.2.3 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the offsite power system included within the BOP.

2. For each BOP system, requirements for offsite a.c. power system:

a. Steady-state load,

b. Inrush kVA for motor loads,

c. Nominal voltage,

d. Allowable voltage regulation,

e. Nominal frequency,

f. Allowable frequency fluctuation,

g. Maximum frequency decay rate and limiting underfrequency value for reactor coolant pump coastdown, and

h. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

NOTE: For complete offsite a.c. power requirements, the BOP designer should also include the NSSS requirements.

8.3 Onsite Power System

8.3.1 A.C. Power Systems

8.3.1.5 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the onsite a.c. power systems included within the BOP.

2. For each BOP system, requirements for onsite a.c. power:

   a. Steady-state load,

   b. Inrush kVA for motor loads,

   c. Nominal voltage,

   d. Allowable voltage drop (to achieve full functional capability within required time period),

   e. Load sequence,

   f. Nominal frequency, and

   g. Allowable frequency fluctuation.

NOTE A: For complete onsite a.c. power requirements, the BOP designer should also include the NSSS requirements.

NOTE B: This interface assumes that the onsite a.c. diesel generator system is utility-specific.

3. Coordination of the design of the diesel generator room with the utility-applicant.

8.3.2 D.C. Power Systems

8.3.2.3 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the d.c. power systems included within the BOP.

2. Provisions included to accommodate the needs of the SSWS (and additional cooling capacity, if required) by the d.c. power systems:

   a. Steady-state load,

   b. Surge loads,

   c. Load sequence,

   d. Nominal voltage, and

   e. Allowable voltage drop (to achieve full functional capability within required time period).

9. AUXILIARY SYSTEMS

9.2 Water Systems

9.2.7 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the SSWS included within the BOP.

2. Integrated heat load (decay heat and station heat load for all NSSS and BOP systems, as a function of time for the various modes of plant operation and limiting accident conditions) that must be transferred to the ultimate heat sink, maximum and minimum temperature limits, pressure, flow rate, plant SSWS pressure drop, etc.

3. Coolant flow, pressure, temperature, and integrated condensate storage capacity to satisfy total plant needs during normal operation, shutdown, and accident conditions. Cooling water requirements for the diesel generator system should be coordinated with the utility-applicant.

4. Limits on quality of makeup water to the station, including conductivity, pH, oxygen, chlorides, fluorides, solids, carbon dioxide, particulates, and silica; and limits on makeup waterflow, temperature, and pressure.

5. Requirements for location and arrangement of potable and sanitary water systems to preclude adverse effects on safety systems and components in the event of failure.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.6 BOP Interface

Ventilation requirements for the diesel generator rooms should be coordinated with the utility-applicant.

9.5 Other Auxiliary Systems

9.5.9 BOP Interface

Site-related requirements to satisfy the fire protection program.

11. RADIOACTIVE WASTE MANAGEMENT

11.2 Liquid Waste Management Systems

11.2.4 BOP Interface

Expected release rates of radioactive material to the environment from the LWMS.

11.3 Gaseous Waste Management Systems

11.3.4 BOP Interface

Expected release rates of radioactive materials to the environment from the GWMS and from other release points including:

1. Location of all release points,
2. Height above grade,
3. Height relative to adjacent buildings,
4. Effluent temperature,
5. Effluent flow rate,
6. Effluent velocity, and
7. Size and shape of flow orifice.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.3 Effluent Monitoring and Sampling

11.5.3.X BOP Interface. Requirements for offsite sampling and monitoring of effluent concentrations.

13. CONDUCT OF OPERATIONS

13.3 Emergency Planning

13.3.3 BOP Interface

Features of the plant that may affect plans for coping with emergencies, as specified in Appendix O to 10 CFR Part 50. Examples of such features are the design of the onsite emergency first aid and personnel decontamination facilities and the emergency operations center; and facility features, including communications systems, that ensure the capability for plant evacuation and reentry (i.e., to mitigate the consequences of an accident or, if appropriate, to continue operations).

13.6 Industrial Security

13.6.3 BOP Interface*

Identification of vital equipment, as defined in 10 CFR §73.2, for use in developing physical security plans. Also, where applicable, locations of items of vital equipment and provisions incorporated into the design for monitoring the status of vital equipment to detect malevolent acts to impair performance.

14. INITIAL TEST PROGRAM

14.1 Specific Information To Be Included in Preliminary Safety Analysis Reports

14.1.8 BOP Interface

Identification of the special or unique features of the NSSS and BOP designs for consideration in developing the initial test program by the utility.

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*This information should be submitted as proprietary information [10 CFR §2.790 (d)].

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