U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

6.5.3 FISSION PRODUCT CONTROL SYSTEMS AND STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of reactor accident consequence assessment, specifically design basis containment and ventilation performance

Secondary - Organization responsible for structural design of containment and systems

Organization responsible for radiation protection

I. AREAS OF REVIEW

The description of the fission product control systems and structures is reviewed to (a) provide a basis for developing the mathematical model for design basis loss-of-coolant accident (LOCA) dose computations, (b) verify that the values of certain key parameters are within pre-established limits, (c) confirm the applicability of important modeling assumptions, and (d) verify the functional capability of ventilation systems used to control fission product releases. The parameters that must be established for use in the calculation of the radiological consequences of accidents in Chapter 15 of the safety evaluation report (SER) and the systems whose functions must be reviewed are outlined below. Many of these areas are the responsibility of staff and are reviewed by the organization responsible for the review of reactor accident consequence assessment, specifically design basis containment and ventilation performance, to provide a general knowledge of the containment systems and their operation following a LOCA.

Revision 3 - March 2007

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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The specific areas of review are as follows:

1. Primary Containment Design

   A. Primary containment characteristics, including (1) the containment isolation times and methods; (2) leak rates prior to and following containment isolation if venting, vacuum relief, or purging of the containment is permitted (by technical specification) during operation; (3) total and mixing volumes to be assumed from the recirculation characteristics given in safety analysis reports (SARs); and (4) the efficiencies of the engineered safety features (ESF) filters used for postaccident ventilation.

      Each of the foregoing containment design and operational characteristics will influence the quantity of radioactive fission products available for release during normal operation and as a consequence of accidents.

   B. The dose mitigating function of the pressure suppression devices e.g. subatmospheric operation, suppression pools, or containment heat removal is described in Sections 6.2.1, 6.2.1.1.A, 6.2.1.1.B, 6.2.1.1.C, and 6.2.2 of the SAR.

      The performance of pressure suppression devices could affect fission product transport and release, as well as containment pressure and containment leakage rate.

2. Secondary Containment Design

   A. Containment type - e.g., metal siding, reinforced concrete (see SAR Section 3.8.4).

      The type of secondary containment structure may affect the potential for exfiltration and the probable leak tightness of the secondary containment.

   B. Physical layout- e.g., volume completely surrounding primary containment, auxiliary building regions treated, main steam tunnel treated (boiling water reactors, BWRs), main steam line leakage control system (BWRs), drawings or plan views defining secondary containment boundary, clarification of which regions are served by cleanup systems (see SAR Sections 6.2.3, 6.5.3, and 9.4).

      Knowledge of what regions are included as part of the secondary containment is essential to establish the mathematical model for dose calculations.

   C. Fission product removal or holdup system design- e.g., regions served by each system, piping and instrumentation drawings of each system and its operation, fan flow rates, recirculation rate, filter locations and efficiencies, system redundancy, actuation signals, time to reduce region pressures below atmospheric, placement of ducting (see SAR Sections 6.2.3, 6.5.1, and 6.5.3).

The reviewer is responsible for determining that each system can perform its functions as claimed to reduce fission product release following a postulated design basis accident. Knowledge of fission products removal systems is necessary for modeling the system for the dose calculation.

D. General design characteristics- e.g., negative pressure during normal operation, free volumes and mixing regions, and leakage rates (see SAR Sections 6.2.3, 6.5.3, and 9.4).

Knowledge of these parameters is necessary for developing the mathematical model for dose calculations.

3. Other Fission Product Control Systems. The ventilation systems used to control fission products and the capability to maintain a negative pressure during accident conditions (see SAR Sections 9.4.1 through 9.4.5).

4. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

5. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. Review of the containment pressure response and mixing fractions, positive pressure periods, and containment leakage rates is performed under SRP 6.2.1 and 6.2.6.

2. Review of the pressure transient in the secondary containment to verify secondary containment region pressures after a design basis accident and to review bypass leakage paths is performed under SRP 6.2.3.

3. Review of the seismic design and quality group classifications for ventilation systems is performed under SRP 3.2.1 and 3.2.2.

4. Review of the functional capability of the filter design associated with the exhaust system is performed under SRP 6.5.1.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. General Design Criterion (GDC) 41 as it relates to the containment atmosphere cleanup system being designed to control fission product releases to the environment following postulated accidents.

2. General Design Criterion (GDC) 42 as it relates to the containment atmosphere cleanup system being designed to permit periodic inspections.

3. General Design Criterion (GDC) 43 as it relates to the containment atmosphere cleanup system being designed to permit appropriate functional testing.

4. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

5. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. Primary Containment. Primary containment design leakage rates for which credit is given should not be less than 0.1% per day due to difficulties in measuring lower leakage rates. Containment isolation methods and times must be such that the calculated radiological doses resulting from the escape of radioactive material prior to and following isolation after a LOCA do not exceed the applicable dose requirements of 10 CFR Part 100 and GDC 19.

   The primary reactor containment and associated systems should be designed so that periodic inspections and functional testing can be performed.

2. Secondary Containment. To be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and at least should be held at a pressure of 0.6 cm (0.25 in) (water), below adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to ensure site boundary exposures less than those calculated for the design basis accidents even if exfiltration occurs.

   Acceptance of other fission product control structures for collection and control of postaccident releases will be determined following consultation with the organization responsible for the review of reactor accident consequence assessment, (specifically design basis containment and ventilation performance) and the organization responsible for structural design of containment and ventilation systems, on a case-by-case basis. The leakage and filtration rates of such structures are acceptable provided that the offsite doses calculated by the organization responsible for radiation protection under SRP Section 15.6.5 will meet the dose guidelines of 10 CFR Part 100 and provided that the preoperational testing and appropriate technical specifications are acceptable.

   Other criteria include specifications for intake and return headers on recirculation systems. These should be placed as far away from each other as practical. The return header should provide a wide distribution over the secondary containment. The purpose of this placement is to ensure some degree of mixing of the return flow in the secondary containment volume before it is again drawn into the system intake.

   With judicious placement, up to 50% mixing may be assumed. A claim for greater than 50% mixing must be supported by the applicant to the satisfaction of the staff. Spacing between intake and return headers is reviewed on a case-by-case basis. Adjustments in the mixing fraction to less than 50% may be indicated by some designs. Past practice has been to allow mixing in 50% of the volume between — and within 3 or 6 meters (10 or 20 feet) of — the inlet and outlet headers if both have distributed openings or if one has distributed openings and the other is at the top of the containment.

   Partial dual containments should meet the same basic criteria as secondary containments in order to be given credit for fission product holdup and removal. The fraction of leakage source considered to be controlled by such partial fission products control structures is determined after consultation with the SCSB reviewer on a case-by-case basis.

4. Other Fission Product Control Systems. Fission product retention credit may be taken by the applicant for other systems- e.g., containment spray systems as evaluated in SRP Section 6.5.2, pressure suppression pools as evaluated in SRP Section 6.5.5, and

filtration and adsorption units as described in Regulatory Guide 1.52. Justification for fission product retention systems should include analytical bases addressing the important physical and chemical variables of the fission product removal and retention processes.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 41 requires that fission product control systems be provided in the reactor containment to reduce the concentration of fission products released to the environment after postulated accidents.

   In the primary containment, fission product control systems include spray, filtration systems, and pressure suppression devices; in the secondary containment, such systems include fission product removal and holdup systems. The function of the fission product control systems is to mitigate the radiological offsite consequences of postulated accidents by decreasing the concentration of fission products available for release to the environment. The review and evaluation of these fission product control systems is the subject of SRP Section 6.5.3. The system and component design criteria for fission product control systems are outlined in Regulatory Guide 1.52, Regulatory Positions C.1, C.2, and C.3.

   Meeting the requirements imposed by GDC 41 provides assurance that offsite radiation doses resulting from postulated accidents will be within the doses specified in 10 CFR Part 100.

2. GDC 42 requires that the fission product control systems be designed to permit periodic inspections of important components.

   The fission product control systems are provided to ensure that offsite radiation doses resulting from postulated accidents are within the doses specified in 10 CFR Part 100. The ability to perform periodic inspection is essential for ensuring that components of the systems will function as designed. Testing and inspection criteria for fission product control systems are outlined in Regulatory Guide 1.52, Regulatory Positions C.5 and C.6.

   Meeting the requirements imposed by GDC 42 provides assurance that offsite radiation doses resulting from postulated accidents will be within the doses specified in 10 CFR Part 100.

3. GDC 43 requires that fission product control systems be designed to permit periodic functional testing of important components.

   Fission product control systems are provided to ensure that offsite radiation doses resulting from postulated accidents are within the doses specified in 10 CFR Part 100. Periodic functional testing is essential for ensuring that components of the systems will function as designed. Testing and inspection criteria for fission product control systems are outlined in Regulatory Guide 1.52, Regulatory Positions C.5 and C.6.

Meeting the requirements imposed by GDC 43 provides assurance that offsite radiation doses resulting from postulated accidents will be within the doses specified in 10 CFR Part 100.

III. REVIEW PROCEDURES

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

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Primary Containment Design

   A. The primary containment design is studied to familiarize the reviewer with the overall construction and anticipated performance capability of the primary containment. Certain parameters and design features, such as design leakage rate, purge/vent systems leakage rate prior to containment isolation, containment free volume, and internal fission product cleanup systems, should be noted for later use (see example of worksheet, Table 6.5.3-1). The performance capability of the internal fission product cleanup systems (if any) should be verified (see SAR Sections 6.5.1, 6.5.2, and 6.5.4).

   B. The transient response of the containment pressure following the accident should be studied. Historically, pressurized water reactor (PWR) containment design leakage rates have been reduced by a factor of two by one day into the accident (Regulatory Guide 1.4), whereas BWR containment design leakage rates were assumed to be constant for all time periods following the accident (Regulatory Guide 1.3[1]). The reviewer should verify with the organization responsible for the review of reactor accident consequence assessment, specifically design basis containment and ventilation performance that these modeling assumptions are valid for each case reviewed. For those containments designed to reach subatmospheric pressure at some time less than 30 days after the accident, the same organization should verify the time necessary to reach subatmospheric pressure.

2. Secondary Containment Design

   A. The design of the secondary containment is reviewed to determine how it should be modeled for the dose calculations. The reviewer also ascertains that the

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[1]Regulatory Guides 1.3 and 1.4 provide guidance related to Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." This guidance is applicable to a holder of an operating license issued prior to January 10, 1997 or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997. These license holders may voluntarily revise the accident source term. Regulatory Guide1.183 is applicable to applicants or license holders issued after January 10, 1997.

applicant has considered the question of potential exfiltration from regions of the secondary containment under varying wind conditions, especially if the structure has a leakage rate greater than 100% per day. The anticipated leakage rate from each region is noted (see example of worksheet, Table 6.5.3-2), and special attention is paid to the accuracy of the proposed leakage testing if the leakage rates are less than 10% per day. (No facility reviewed to date has a proposed secondary containment leakage rate of less than 10% per day. Experience indicates that 10% per day may be difficult to achieve in actual practice.)

B. The boundary of the secondary containment is determined. Usually, the secondary containment boundary is composed of more than one region e.g., a shield building (concrete) or enclosure building (metal siding) around the primary containment and all or parts (emergency core cooling pump rooms, etc.) of the auxiliary building. These regions may be serviced by one or more ventilation systems.

3. Other Fission Product Control Systems

A. For PWR containments and BWR MARK III containments, the annular region between the shield building or enclosure building and the primary containment may be held at a negative pressure relative to adjacent areas by a vacuum exhaust system during normal operation. Since this system is used during normal operation, it may appear in the SAR under auxiliary systems. The exhaust system may also treat the auxiliary building regions that are part of the secondary containment, but if these regions are maintained at a negative pressure during normal operation, it is most likely done with the auxiliary building ventilation system. The ability of the vacuum exhaust and auxiliary building ventilation systems falls under the purview of the organization responsible for the review of reactor accident consequence assessment, specifically design basis containment and ventilation performance. The systems' ability to maintain a negative pressure of sufficient margin under varying wind conditions and operational modes prior to a design basis accident is verified by the same organization. The reviewer also verifies that the systems have the capability to maintain negative pressure following a design basis accident. If an adequate negative differential pressure (0.6 cm, or 0.25 in, water gauge) is achieved within 60 seconds from the time the accident, then no positive pressure time period need be assumed in the dose model. All positive pressure periods at any time in the secondary containment regions are treated as direct outleakage periods following an accident, and no credit is given for filters or recirculation systems. The organization responsible for the review of reactor accident consequence assessment (specifically design basis containment and ventilation performance) verifies the positive pressure periods. The large reactor buildings around older BWR containments are usually maintained at a negative pressure during normal operation, and the dose model used for these cases has not assumed any positive pressure period.

B. The exhaust systems used to maintain the negative pressure differential following the accident should be sized to meet the negative pressure criterion for the inleakage rate and the conservatively calculated heat load for the regions served by each rate, and analyses to this effect should be presented by the applicant. The pressure response analyses are reviewed by the organization responsible for the review of reactor accident consequence assessment, specifically design basis containment and ventilation performance. The functional capability of the filter design associated with the exhaust system is reviewed by the organization responsible for radiation protection under SRP Section 6.5.1. Design guidance for postaccident cleanup air filtration and adsorption systems is provided in Regulatory Guide 1.52. The reviewer should consult with the organization responsible for radiation protection concerning filter system efficiencies. The exhaust systems may be one of several designs. Common designs are:

i. Straight exhaust through charcoal and high-efficiency particulate air (HEPA) filters. Primary containment leakage to these regions is assumed to go directly to the filter with no mixing or holdup in the region being filtered.

ii. Recirculation system with split inflow (some exhausted through filters and some recirculated to the region being served). Primary containment leakage to the region being serviced is assumed to occur directly to the intake of the recirculation fan. There, a fraction of the leakage (the ratio of exhaust to total flow) is exhausted through the filters; the balance is then assumed to return to the region being serviced. The placement of the system intake and return headers is examined to determine that return flow from the fans does not have a direct path to the intake again. Credit for mixing in 50% of the region is given for fission products returned by the recirculation system to the secondary volume if the header placement is satisfactory.

iii. Other variations on the recirculation system are (a) filters in the recirculation line, (b) filters in both the recirculation line and the exhaust line, and (c) high exhaust flow to reduce the negative pressure to several centimeters (inches) water gauge, and then no exhaust with recirculation only for some time period.

C. The sizing of the system fans for the volumes they are maintaining at a negative pressure may be critical in determining the ratio of exhaust flow to recirculation flow. Past history shows that secondary containment structures are considerably more leaky than applicants anticipated (2 to 5 times as great as anticipated), and fan exhaust flows have been increased after testing to account for this. (When identical flow rates are predicted for two volumes that differ by a factor of 10 or more, it is difficult to believe that the negative pressure differential will be the same for both volumes.) The flow rates, negative pressure differential, and volumes are noted, and the organization responsible for the review of reactor accident consequence assessment (specifically design basis containment and ventilation performance, pressure response only) is consulted for verification before performing dose calculations.

D. The systems should be reviewed to determine volumes exhausted, system operation, fan flow rates, and filter efficiencies. All the applicant's claims should be verified by appropriate staff members as noted in Table 6.5.3-2 of this SRP section. Leakage fractions from the primary containment to each volume should be identified and stated in the technical specifications. Completeness of information, adequacy of technical specifications and testing methods, and the adequacy and maintenance of the integrity of the secondary containment negative pressure considering failures of nonseismic piping or ducting are verified by the organization responsible for the review of reactor accident consequence assessment, specifically design basis containment and ventilation performance.

4. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

   For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

1. In sum, the fission product control systems and structures for mitigation of offsite doses resulting from design basis LOCA have been reviewed. The review has included the applicant's proposed design criteria and design bases for each system and the applicant's analysis of the adequacy of those criteria and bases. The applicant's analyses of the manner in which the designs of the fission product control systems conform to the proposed design criteria have also been reviewed.

2. The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the fission product control systems and necessary auxiliary supporting systems to the Commission's regulations as outlined in 10 CFR Part 50, Appendix A, General Design Criteria 41, 42, and 43, and to the guidance of staff technical positions and industry standards.

3. The applicant's design of the fission product control systems has been reviewed to ensure that the parameters presented in Tables 6.5.3-1 and 6.5.3-2 are appropriate for calculation of the post-LOCA doses as outlined in SRP Section 15.6.5.

4. Accordingly, the staff concludes that the fission product control systems and structures are acceptable and meet the relevant requirements of General Design Criteria 41, 42, and 43.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

2. 10 CFR Part 50, Appendix A, General Design Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems."

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems."

4. 10 CFR Part 100, "Reactor Site Criteria."

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

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

7. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption."

8. NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections," issued April 25, 2006.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

Units of Light-Water-Cooled Nuclear Power Plants
Table 6.5.3-1

Primary Containment Parameters

Data Description	Parameter Value
Type of Structure	
Primary Containment Design Leak Rate	
Bypass Leakage Fraction to Volumes	
1.	
2.	
3.	
Primary Containment Free Volume	
Primary Containment Subatmospheric Operation	
Primary Containment Internal Fission Product Removal Systems:	
Filter System	
Other	
Primary Containment Purge/Vent Operation:	
Leakage During Normal Operation	
Valve Arrangement	
Accident Leakage Via Purge/Vent System Prior to Containment Isolation	

Table 6.5.3-2

Secondary Containment Parameters

Data Description	Parameter Value
For each Secondary Containment Region:	
Type of Structure	
Free Volume	
Mixing Fraction	
Design Leak Rate	
Annulus Width (where applicable)	
For each Ventilation System:	
Total Recirculation Flow	
Exhaust Flow	
Filter Placement	
Filter Efficiencies	
Header Placement	
Time Sequence for Operation Following an Accident or	
Operation of System Prior to an Accident if Used During Normal Operation	