UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 31, 1994

Mr. Richard Ochs, Director
Maryland Safe Energy Coalition
P.O. Box 33111
Baltimore, MD 21218

SUBJECT: ISOLATION PROVISIONS FOR THE SERVICE WATER SYSTEM - CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M87189) AND UNIT NO. 2 (TAC NO. M87190)

Dear Mr. Ochs:

I am responding to your letter dated June 27, 1995. You indicated that the Maryland Safe Energy Coalition is concerned that some issues are not being addressed concerning seismic hazards and remedies. You indicated that nearby residents of the plant have heard rumbles, explosions, crashes, booms,and bangs from the Calvert Cliffs Nuclear Power Plant. I am in contact with the resident inspectors and the licensee on a daily basis and neither they nor I are aware of such occurrences except for an occasional lifting of the safety valves (SVs) or the atmospheric steam dump valves (ASDVs). The SVs have only lifted approximately 5 times and the ASDVs approximately 11 times in the last 5 years. The lifting of the SVs or the ASDVs usually occur for only a few seconds accompanied by a loud noise; however, I don't believe that tremors or vibrations can be felt at locations adjacent to the power plant property due to the lifting of the SVs or the ASDVs.

The ASDVs are located on the secondary steam system and are available to relieve steam pressure if there is a turbine trip or loss-of-condenser vacuum which does not allow the turbine bypass system to perform this function.

"Codes and Standards," 10 CFR 50.55a, requires that pressure-retaining components of the reactor coolant pressure boundary (RCPB), designated Class 1, be designed and constructed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III. The ASME Code further requires overpressure protection for the RCPB for normal, abnormal, and accident conditions. The SVs lifted as the result of abnormal operational transients and resultant reactor scrams in accordance with their design basis.

The lifting of the SVs or the ASDVs is not considered a seismic event and are not recorded.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (SRP) dated June 1978, identifies the NRC staff's review process including the areas reviewed, acceptance criteria, review procedures, and evaluation findings. Although the SRP was issued subsequent to the licensing of the Calvert Cliffs facility, it documented the NRC staff's licensing review process that was being used for plants being licensed in the early 1970s, such as Calvert Cliffs. The SRP was developed to inform licensees and the public of the NRC's review process. The following SRP sections indicate how the NRC staff addressees your concerns relating to the amplitude, duration, number, and natural frequencies of transients during normal, abnormal, and accident conditions including the resultant impact of the transients on structures, systems, or components:

Section 3.7.3, "Seismic Subsystem Analysis." (Enclosure 1)
Section 3.7.4, "Seismic Instrumentation." (Enclosure 2)
Section 3.9.1, "Special Topics for Mechanical Components." (Enclosure 3)
Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment." (Enclosure 4)
Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures." (Enclosure 5)
Section 3.9.6, "Inservice Testing of Pumps and Valves." (Enclosure 6)
Section 6.6, "Inservice Inspection of Class 2 and 3 Components." (Enclosure 7)
Section 10.3, "Main Steam Supply System" (Enclosure 8)

As detailed in the SRP, the NRC staff considers the impact of transients for normal, abnormal, and accident conditions as well as external events, such as seismic, during the licensing and operational life of nuclear power plants. The structures, systems, and components are designed, analyzed, and tested to assure that they will meet their safety-related functions under all anticipated conditions.

Pre-operational testing is performed during startup to confirm that the design specifications are met and surveillance testing, inservice testing, and inservice inspections are performed during plant operation or shutdown to assure that design and safety margins are maintained during the licensed life of a nuclear power plant. The SRP is available in the Public Document Room which has more detailed information relative to the design requirements and NRC staff review process.

As noted in our request for additional information which you referenced, we requested additional information on the ranges of natural frequencies of the non-safety portion of the service water piping, the basis for the selection of the postulated break locations, and the basis for concluding that the integrity of the turbine building would be maintained during a seismic event. We will determine the adequacy of the licensee's interim actions until its completion of the Examination of External Events (IPEEE) program for the Calvert Cliffs site. The IPEEE is being performed by all nuclear power plants, on a site specific basis, to determine if any additional actions need to be taken to provide adequate protection from external events. If, during this interim period, we or the licensee identify any safety concerns in this area that might impact the NRC staff's conclusions concerning the acceptability of the existing service water system configuration, we will take whatever action is necessary to assure the continued safe operation of the Calvert Cliffs Nuclear Power Plant.

I hope this has been responsive to your concerns.

Sincerely,

Daniel G. McDonald

Daniel G. McDonald, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-317
and 50-318

Enclosures: As stated

cc w/encls: See next page

Robert E. Denton
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acceptability of the existing service water system configuration, we will take whatever action is necessary to assure the continued safe operation of the Calvert Cliffs Nuclear Power Plant.

I hope this has been responsive to your concerns.

Sincerely,

Original signed by:
Daniel G. McDonald, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-317
and 50-318

Enclosures: As stated

cc w/encls: See next page

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SECTION 3.7.3 SEISMIC SUBSYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Structural and Geosciences Branch (ESGB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic subsystem analysis are reviewed:

1. Seismic Analysis Methods

The information reviewed is similar to that described in subsection I.1 of Standard Review Plan (SRP) Section 3.7.2 but as applied to seismic Category I subsystems.

2. Determination of Number of Earthquake Cycles

Criteria or procedures used to establish the number of earthquake cycles during one seismic event and the maximum number of cycles for which applicable Category I subsystems and components are designed are reviewed.

3. Procedures Used for Analytical Modeling

The criteria and procedures used for modeling the seismic subsystems are reviewed.

4. Basis for Selection of Frequencies

As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed.

Enclosure 1

5. Analysis Procedure for Damping

The information reviewed is similar to that described in subsection I.13 of SRP Section 3.7.2 but as applied to Category I subsystems.

6. Three Components of Earthquake Motion

The information reviewed is similar to that described in subsection I.6 of SRP Section 3.7.2 but as applied to Category I subsystems.

7. Combination of Model Responses

The information reviewed is similar to that described in subsection I.7 of SRP Section 3.7.2 but as applied to Category I subsystems.

8. Interaction of Other Systems With Category I Systems

The seismic analysis procedures to account for the seismic motion of non-Category I systems in the seismic design of Category I systems are reviewed.

9. Multiply-Supported Equipment and Components with Distinct Inputs

The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are reviewed.

10. Use of Equivalent Vertical Static Factors

The information reviewed is similar to that described in subsection 1.10 of SRP Section 3.7.2 but as applied to Category I subsystems.

11. Torsional Effects of Eccentric Masses

The criteria and procedures that are used to consider the torsional effects of eccentric masses in seismic subsystem analyses are reviewed.

12. Category I Buried Piping, Conduits, and Tunnels

For Category I buried piping, conduits, tunnels, and auxiliary systems, the seismic criteria and methods which consider the compliance characteristics of soil media, dynamic pressures, settlement due to earthquake and differential movements at support points, penetrations, and entry points into structures provided with anchors are reviewed.

13. Methods for Seismic Analysis of Category I Concrete Dams

The analytical methods and procedures that will be used for seismic analysis of Category I concrete dams are reviewed. The assumptions made, the boundary conditions used, the hydrodynamic effects considered, and the procedures by which strain-dependent material properties of foundation are incorporated in the analysis are reviewed.

14. Methods for Seismic Analysis of Above-Ground Tanks

For Category I above-ground tanks, the seismic criteria and methods that consider hydrodynamic forces, tank flexibility, soil-structure interaction, and other pertinent parameters are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in subsection I of this SRP section are given below. Other criteria which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used. The staff accepts the design of subsystems that are important to safety and must withstand the effects of earthquakes if the relevant requirements of General Design Criterion (GDC) 2 (Ref. 1) and Appendix A to 10 CFR Part 100 (Ref. 2) concerning material phenomena are complied with. The relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100 are:

1. General Design Criterion 2 - The design basis shall reflect appropriate consideration of the most severe earthquakes reported to have affected the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.

2. Appendix A to 10 CFR Part 100 - Two earthquake levels, the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE), shall be considered in the design of safety-related structures, components, and systems. Appendix A to 10 CFR Part 100 further states that the design used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the safe shutdown earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.

Specific criteria necessary to meet the relevant requirements of GDC 2 and Appendix A to Part 100 are as follows:

1. Seismic Analysis Methods

   The acceptance criteria provided in SRP Section 3.7.2, subsection II.1, are applicable.

2. Determination of Number of Earthquake Cycles

   During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBEs) should be assumed. The number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed.

3. Procedures Used for Analytical Modeling

   The acceptance criteria provided in SRP Section 3.7.2, subsection II.3, are applicable.

4. Basis for Selection of Frequencies

To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.

5. Analysis Procedure for Damping

The acceptance criteria provided in SRP Section 3.7.2, subsection II.13, are applicable.

6. Three Components of Earthquake Motion

The acceptance criteria provided in SRP Section 3.7.2, subsection II.6, are applicable.

7. Combination of Modal Responses

The acceptance criteria provided in SRP Section 3.7.2, subsection II.7, are applicable.

8. Interaction of Other Systems With Category I Systems

To be acceptable, each non-Category I system should be designed to be isolated from any Category I system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I system. If it is not feasible or practical to isolate the Category I system, adjacent non-Category I systems should be analyzed according to the same seismic criteria as applicable to the Category I system. For non-Category I systems attached to Category I systems, the dynamic effects of the non-Category I systems should be simulated in the modeling of the Category I system. The attached non-Category I systems, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I system.

9. Multiply-Supported Equipment and Components With Distinct Inputs

Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, by using the floor response spectra. For the latter option the maximum displacement of each support is predicted by $S_d = S_a g/\omega^2$, where S_a is the spectral acceleration in "g's" at the high-frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method.

In the case of multiple supports located in a single structure, an alternative acceptable method using the floor response spectra involves determination of dynamic responses due to the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors and applied identically to all the floors, provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination using static analysis procedures.

In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the subsystems. Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiply-supported systems.

10. Use of Equivalent Vertical Static Factors

The acceptance criteria provided in SRP Section 3.7.2, subsection II.10, are applicable.

11. Torsional Effects of Eccentric Masses

For seismic Category I subsystems, when the torsional effect of an eccentric mass is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for judging the significance will be reviewed on a case-by-case basis.

12. Category I Buried Piping, Conduits, and Tunnels

For Category I buried piping, conduits, tunnels, and auxiliary systems, the following items should be considered in the analysis:

a. Two types of groundshaking-induced loadings must be considered for design.

(i) Relative deformations imposed by seismic waves traveling through the surrounding soil or by differential deformations between the soil and anchor points.

(ii) Lateral earth pressures and ground-water effects acting on structures.

b. The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Procedures using the principles of the theory of structures on elastic foundations are acceptable.

c. When applicable, the effects due to local soil settlements, soil arching, etc., should also be considered in the analysis.

d. Actual methods used for determining the design parameters associated with seismically induced transient relative deformations are reviewed and accepted on a case-by-case basis. Additional information, for guidance purposes only, can be found on page 26 of Reference 3 and in Section 3.5.2 of Reference 4.

13. Methods for Seismic Analysis of Category I Concrete Dams

For the analysis of all Category I concrete dams, an appropriate approach that takes into consideration the dynamic nature of forces (due to both horizontal and vertical earthquake loadings), the behavior of the dam material under earthquake loadings, soil-structure interaction (SSI) effects, and nonlinear stress-strain relations for the soil, should be used. Analysis of earthen dams is reviewed under Section 2.5.6.

14. Methods for Seismic Analysis of Above-Ground Tanks

Most above-ground fluid-containing vertical tanks do not warrant sophisticated, finite element, fluid-structure interaction analyses for seismic loading. However, the commonly used alternative of analyzing such tanks by the "Housner-method" (Ref. 5) may be inadequate in some cases. The major problem is that direct application of this method is consistent with the assumption that the combined fluid-tank system in the horizontal impulsive mode is sufficiently rigid to justify the assumption of a rigid tank. For flat-bottomed tanks mounted directly on their bases, or tanks with very stiff skirt supports, the assumption leads to the usage of a spectral acceleration equal to the zero-period base acceleration. Recent studies (Refs. 6, 7, 8, 9, and 10) have shown that for typical tank designs the frequency for this fundamental horizontal impulsive mode of the tank shell and contained fluid is such that the spectral acceleration may be significantly far greater than the zero-period acceleration. Thus, the assumption of a rigid tank could lead to inadequate design loadings. The SSI effects may also be very important for tank responses, and they may be considered for both horizontal and vertical motions.

The acceptance criteria below are based upon the information contained in References 1 through 3 and Reference 5. These references also contain acceptable calculational techniques for the implementation of these criteria. The use of other approaches meeting the intent of these criteria can also be considered if adequate justification is provided.

a. A minimum acceptable analysis must incorporate at least two horizontal modes of combined fluid-tank vibration and at least one vertical mode of fluid vibration. The horizontal response analysis must include at least one impulsive mode in which the response of the tank shell and roof are coupled together with the portion of the fluid contents that moves in unison with the shell. Furthermore, at least the fundamental sloshing (convective) mode of the fluid must be included in the horizontal analysis.

b. The fundamental natural horizontal impulsive mode of vibration of the fluid-tank system must be estimated giving due consideration to the flexibility of the supporting medium and to any uplifting tendencies for the tank. It is unacceptable to assume a rigid tank unless the assumption can be justified. The horizontal impulsive-mode spectral acceleration, S_{a1}, is then determined using this frequency and the appropriate damping for the fluid-tank system. Alternatively, the maximum spectral acceleration corresponding to the relevant damping may be used.

c. Damping values used to determine the spectral acceleration in the impulsive mode shall be based upon the system damping associated with the tank shell material as well as with the SSI, as specified in References 3 and 10.

d. In determining the spectral acceleration in the horizontal convective mode, S_{a2}, the fluid damping ratio shall be 0.5 percent of critical damping unless a higher value can be substantiated by experimental results.

e. The maximum overturning moment, M_o, at the base of the tank should be obtained by the modal and spatial combination methods discussed in subsection II of SRP Section 3.7.2. The uplift tension resulting from M_o must be resisted either by tying the tank to the foundation with anchor bolts, etc., or by mobilizing enough fluid weight on a thickened base skirt plate. The latter method of resisting M_o must be shown to be conservative.

f. The seismically induced hydrodynamic pressures on the tank shell at any level can be determined by the modal and spatial combination methods in SRP Section 3.7.2. The maximum hoop forces in the tank wall must be evaluated with due regard for the contribution of the vertical component of ground shaking. The beneficial effects of soil-structure interaction may be considered in this evaluation (Refs. 4, 11, 12, and 13). The hydrodynamic pressure at any level must be added to the hydrostatic pressure at that level to determine the hoop tension in the tank shell.

g. Either the tank top head must be located at elevation higher than the slosh height above the top of the fluid or else must be designed for pressures resulting from fluid sloshing against this head.

h. At the point of attachment, the tank shell must be designed to withstand the seismic forces imposed by the attached piping. An appropriate analysis must be performed to verify this design.

i. The tank foundation (see also SRP Section 3.8.5) must be designed to accommodate the seismic forces imposed on it. These forces include the hydrodynamic fluid pressures imposed on the base of the tank as well as the tank shell longitudinal compressive and tensile forces resulting from M_o.

j. In addition to the above, a consideration must be given to prevent buckling of tank walls and roof, failure of connecting piping, and sliding of the tank.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case. The review procedures are such as to satisfy the requirements of acceptance criteria stated in subsection II.

1. Seismic Analysis Methods

   The seismic analysis methods are reviewed to determine that these are in accordance with the acceptance criteria of SRP Section 3.7.2, subsection II.1.

2. Determination of Number of Earthquake Cycles

   Criteria or procedures used to establish the number of earthquake cycles are reviewed to determine that they are in accordance with the acceptance criteria as given in subsection II.2 of this SRP section. Justification for deviating from the acceptance criteria is requested from the applicant, as necessary.

3. Procedures Used for Analytical Modeling

   The criteria and procedures used for modeling for the seismic subsystem analysis are reviewed to determine that these are in accordance with the acceptance criteria of SRP Section 3.7.2, subsection II.3.

4. Basis for Selection of Frequencies

   As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed to determine compliance with the acceptance criteria of subsection II.4 of this SRP section.

5. Analysis Procedure for Damping

   The analysis procedure to account for damping in different elements of the model of a coupled system is reviewed to determine that it is in accordance with the acceptance criteria of SRP Section 3.7.2, subsection II.13.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of subsystems are reviewed to determine compliance with the acceptance criteria of SRP Section 3.7.2, subsection II.6.

7. Combination of Modal Responses

The procedures for combining modal responses are reviewed to determine compliance with the acceptance criteria of SRP Section 3.7.2, subsection II.7 when a response spectrum modal analysis method is used.

8. Interaction of Other Systems with Category I Systems

The criteria used to design the interfaces between Category I and non-Category I systems are reviewed to determine compliance with the acceptance criteria of subsection 11.8 of this SRP section.

9. Multiply-Supported Equipment and Components With Distinct Inputs

The criteria for the seismic analysis of multiply-supported equipment and components with distinct inputs are reviewed to determine that the criteria are in accordance with the acceptance criteria of subsection II.9 of this SRP section.

10. Use of Equivalent Vertical Static Factors

Use of equivalent static factors as response loads in the vertical direction for the seismic design of any Category I subsystems in lieu of a detailed dynamic method is reviewed to determine that constant static factors are used only if the structure is rigid in the vertical direction.

11. Torsional Effects of Eccentric Masses

The procedures for seismic analysis of Category I subsystems are reviewed to determine compliance with the acceptance criteria of subsection II.11 of this SRP section.

12. Category I Buried Piping, Conduits, and Tunnels

The analysis procedures for Category I buried piping, conduits, tunnels, and auxiliary systems are reviewed to determine that they are in accordance with the acceptance criteria of subsection II.12 of this SRP section. The analysis includes review of the procedures used to consider the inertial effects of soil media and the differential displacements at structural penetrations, etc. Any procedures that are not adequately justified are so identified, and the applicant is requested to provide additional justification.

13. Methods for Seismic Analysis of Category I Concrete Dams

Methods for the seismic analysis of Category I concrete dams are reviewed to determine compliance with the acceptance criteria of subsection II.13 of this SRP section.

14. Method for Seismic Analysis of Above-Ground Tanks

Methods for seismic analysis of Category I above-ground tanks are reviewed to determine compliance with the acceptance criteria of subsection II.14 of this SRP section.

IV. EVALUATION FINDINGS

Evaluation findings for SRP Section 3.7.3 have been combined with those of SRP Section 3.7.2 and are given under SRP Section 3.7.2, subsection IV.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with the Commission's regulations.

The provisions of this SRP section apply to reviews of construction permit (CP), preliminary design approval (PDA), final design approval (FDA), and combined license (CP/OL) applications docketed after the date of issuance of this SRP section. Operating license (OL) and final design approval (FDA) applications, whose CP and PDA reviews were conducted prior to the issuance of this revision to SRP Section 3.7.3, will be reviewed in accordance with the acceptance criteria given in the SRP Section 3.7.3, Revision 1, dated July 1981.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomenon."

2. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

3. D. W. Coats, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria," NUREG/CR-1161, May 1980.

4. ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers, September 1986.

5. TID-7024, "Nuclear Reactors and Earthquakes," Division of Reactor Development, U.S. Atomic Energy Commission, August 1963.

6. A. S. Veletsos, "Seismic Effects in Flexible Liquid Storage Tanks," Proceedings of Fifth World Conference on Earthquake Engineering, Rome, 1974.

7. A. S. Veletsos and J. Y. Yang, "Earthquake Response of Liquid Storage Tanks," Advances in Civil Engineering Through Engineering Mechanics, Proceedings of the Engineering Mechanics Division Specialty Conference, ASCE, Raleigh, North Carolina, pp. 1-24, 1977.

8. M. A. Haroun and G. W. Housner, "Seismic Design of Liquid Storage Tanks," Journal of the Technical Councils, ASCE, Vol. 107, No. TC1, pp. 191-207, 1981.

9. A. S. Veletsos, "Seismic Response and Design of Liquid Storage Tanks," Guidelines for the Seismic Design of Oil and Gas Pipeline Systems, Technical Council on Lifeline Earthquake Engineering, ASCE, pp. 255-370 and 443-461, 1984.

10. A. S. Veletsos and Y. Tang, "Soil-Structure Interaction Effects for Laterally Excited Liquid-Storage Tanks," to appear as an EPRI Technical Report, Palo Alto, California, 1989.

11. M. A. Haroun and M. A. Tayel, "Axisymmetrical Vibrations of Tanks--Numerical," Journal of Engineering Mechanics Division, ASCE, Vol. 111, No. 3, pp. 329-345, 1985.

12. A. S. Veletsos and Y. Tang, "Dynamics of Vertically Excited Liquid Storage Tanks," Journal of Structural Engineering, ASCE, Vol. 112, No. 6, pp. 1228-1246, 1986.

13. A. S. Veletsos and Y. Tang, "Interaction Effects in Vertically Excited Steel Tanks," Dynamic Response of Structures, G. C. Hart and R. B. Nelson, Editors, ASCE, pp. 636-643, 1986.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

3.7.4 SEISMIC INSTRUMENTATION

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic instrumentation program are reviewed:

1. Comparison with Regulatory Guide 1.12

   A comparison of the proposed seismic instrumentation with the seismic instrumentation guidelines of Regulatory Guide 1.12 (Ref. 4) is made. In addition, the bases for elements of the program that differ from Regulatory Guide 1.12 are reviewed.

2. Location and Description of Instrumentation

   The locations for the installation of seismic instrumentation such as triaxial peak accelerographs, triaxial time history accelerographs, and triaxial response spectrum recorders that will be installed in selected Category I structures and components are reviewed. The bases for selection of the instrumentation and the locations and a discussion of the extent to which the seismic instrumentation will be employed to verify the seismic analyses following an earthquake are reviewed.

3. Control Room Operator Notification

   The procedures that will be followed to inform the control room operator of the peak acceleration level and the input response spectra values shortly after occurrence of an earthquake are reviewed. Also reviewed are the bases for establishing predetermined values for activating the readout of the seismic instrumentation to the control room operator.

Enclosure 2

4. Comparison of Measured and Predicted Responses

The criteria and procedures that will be used to compare measured responses of Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses are reviewed.

5. Inservice Surveillance

The requirements for inservice inspection, testing and calibration as pertaining to operability and reliability are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in subsection I of this SRP section are given below. Any other seismic instrumentation program which can be justified to be equivalent to the acceptance criteria may be used. SEB accepts the seismic instrumentation system if the relevant requirements of General Design Criterion 2 (Ref. 2), 10 CFR Part 100, Appendix A (Ref. 3), and 10 CFR Part 50, § 50.55a (Ref. 1), as they relate to the capabilities and performance of the instruments to adequately measure the effects of earthquakes are met. Specific criteria necessary to meet the requirements of GDC 2, 10 CFR Part 100, Appendix A, and 10 CFR Part 50, § 50.55a, are as follows:

The instrumentation used for the measurements should be capable of recording the effects produced by the most severe earthquakes that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity and period of time in which historical data has been accumulated.

It is required in 10 CFR Part 100, Appendix A, that suitable instrumentation shall be provided so that the seismic response of nuclear plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis.

1. Comparison with Regulatory Guide 1.12

The seismic instrumentation program is considered to be acceptable if it is in accordance with Regulatory Guide 1.12 (see also Table 3.7.4-1). This guide recommends provision of a triaxial time history accelerograph and a triaxial response spectrum recorder to measure the input time history and response spectra directly. Additional time history accelerographs, response spectrum recorders, peak accelerographs, and seismic switches are recommended to measure the responses of structures, equipment, and components at selected locations. The bases for elements of the proposed seismic instrumentation program that differ from Regulatory Guide 1.12 must be provided.

2. Location and Description of Instrumentation

For the construction permit review there should be a commitment by the applicant to provide the following instruments at the given locations:

a. A triaxial time history accelerograph in the free field or at the containment foundation, with readout in the control room.

b. A seismic switch on the containment foundation, with readout in the control room.

c. A triaxial response spectrum recorder on the containment foundation, with readout in the control room.

In addition, a commitment to provide the recommended additional instrumentation at the various response locations should be made without providing details of actual locations.

For the operating license review, a detailed seismic instrumentation plan including details of the locations, mounting and descriptions of the instrumentation should be provided. To be acceptable, the remaining instrumentation locations are related to the locations of the output vibratory motions used in the seismic design. Typical general locations are:

a. Containment structure or reactor building.

b. Reactor piping.

c. Reactor equipment.

d. Other Category I structures, equipment, and piping.

Instrumentation should be provided depending upon the plant safe shutdown earthquake acceleration as given in Regulatory Guide 1.12. The specific locations are determined by the plant designer so as to obtain the most pertinent information. A possible approach to the specification of the seismic instrumentation system is given in Regulatory Guide 1.12. Other desirable combinations of instruments which may prove to be as useful as the instrumentation plan outlined in the guide may be utilized.

The criteria for selection of Category I structures, components, and equipment to be instrumented and the location of instrumentation, as well as the extent to which this instrumentation is employed to verify the seismic analyses following an earthquake, should be specified. The criteria will be reviewed on a case-by-case basis.

3. Control Room Operator Notification

To be acceptable, the seismic switch located at the foundation of the containment should be connected to event indicators that are located in the control room, so that a signal is given when the preset threshold level (OBE acceleration level) resulting from the earthquake is exceeded. Also both audio and visual signals should be provided to the control room operators in the event of an earthquake.

In addition, the triaxial time history accelerograph located in the containment foundation or in the free field should be connected to the control room, so that peak acceleration level experienced in the basement of the reactor containment structure or in the free field is indicated to the control room operator. The response spectrum recorder in the reactor containment foundation or in the free field is also connected to the control room to indicate if the design response spectra values for discrete frequencies are exceeded during an earthquake.

4. Comparison of Measured and Predicted Responses

In the event of an earthquake, the control room operator should be immediately informed through the event indicators. If the instrumentation shows that the peak acceleration or the response spectra experienced at the foundation of the containment building or in the free field exceed the OBE acceleration level or response spectra, the plant should be shut down (Ref. 3) pending permission to resume operations. To help predict the capability of the plant for resuming operations, field inspection of safety-related items should be implemented and the measured responses from both the peak-recording and strong motion accelerographs should be compared with those assumed in the design.

The procedures for comparison of measured and predicted responses are acceptable if a commitment is made to provide detailed comparisons, as outlined below, between measured seismic responses of Category I structures and equipment with calculated responses determined from dynamic analysis. First, the time history records are digitized and corrected for time signal variations and baseline variations. The time history records from the triaxial sensors located in the free field or at the foundation of the containment building are used to calculate response spectra at appropriate critical damping values. The response spectra thus obtained, or the response spectra from the response spectrum recorder, are compared with the design response spectra. In addition, the time history records from the free field triaxial sensor are used as input ground motion for the reactor building dynamic model, including soil where applicable. Amplified response spectra are then calculated at the locations of the other sensors in the reactor building for comparison and correlation with the response spectra directly measured. Structural responses and amplified response spectra are calculated using the free field time history records with the dynamic model for comparison with the original design and analysis parameters. This comparison permits evaluation of seismic effects on structures and equipment and forms the basis for remodeling, detailed analyses, and physical inspection.

5. Inservice Surveillance

Each of the seismic instruments shall be demonstrated operable by the performance of the channel check, channel calibration, and channel functional test operations at the intervals specified in Table 3.7.4-2.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Comparison with Regulatory Guide 1.12

The seismic instrumentation program is checked to assure that the instrumentation is in accordance with the guidelines of Regulatory Guide 1.12.

Any differences between the proposed and the guide seismic instrumentation, which have not been adequately justified, are identified and the applicant is informed of the need for additional technical justification.

2. Location and Description of Instrumentation

At the operating license stage, the locations and descriptions of the seismic instrumentation are reviewed to determine that these are in accordance with the acceptance criteria of subsection II.2 of this SRP section. If the instrumentation provided is judged to be insufficient, the need for additional instrumentation is transmitted to the applicant.

3. Control Room Operator Notification

The seismic instrumentation is checked to verify that the seismic switch located at the foundation of the containment structure or in the free field is connected to event indicators that are located in the control room, so that a signal is given when the preset threshold level is exceeded. If there is no provision for both audio and visual signals in the applicant's seismic instrumentation plan, the applicant is so informed with a request for compliance.

4. Comparison of Measured and Predicted Responses

The criteria and procedures that will be used to compare measured responses of Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses are checked to verify that sufficient information as specified in subsection II.4 of this SRP section is included. Any deficiency in the required information is identified and the applicant is requested to provide further information.

5. Inservice Surveillance

The inservice inspection program described by the applicant is reviewed to assure that the acceptance criteria of subsection II.5 of this SRP section are met.

IV. EVALUATION FINDINGS

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

The staff concludes that the seismic instrumentation system provided for the plant is acceptable and meets the requirements of General Design Criterion 2, 10 CFR Part 100, Appendix A and 10 CFR Part 50, § 50.55a. This conclusion is based on the following:

The applicant has met the requirements of 10 CFR Part 100, Appendix A by providing the instrumentation that is capable of measuring the effects of an earthquake which meets the requirements of GDC 2. The applicant has met the requirements of 10 CFR Part 50, § 50.55a by providing the inservice inspection program that will verify operability by performing channel checks, calibrations, and functional tests at acceptable intervals. In addition, the installation of the specified seismic instrumentation in the reactor containment structure and at other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

VI. REFERENCES

1. 10 CFR Part 50, § 50.55a "Codes and Standards."

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

3. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

4. Regulatory Guide 1.12, "Instrumentation for Earthquakes."

TABLE 3.7.4-1 SEISMIC INSTRUMENTATION REQUIREMENTS

Instrumentation	Triaxial Time-History Accelerograph		Triaxial Response Spectrum Recorder		Triaxial Peak Accelerograph		Seismic Switch	
Location SSE	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g
I. Free Field	1*#	1*#						
II. Inside Containment								
Basement	1*	1*	1*	1*			1*	1*
At Elevation	1	1						
Reactor Equip. Sup.			}1	}1				}1*
Reactor Piping Sup.								
Reactor Equipment					1	1		
Reactor Piping					1	1		
III. Outside Containment								
Cat. I Structure		1	1	1				
Cat. I Equip. Sup.				1				
Cat. I Piping Sup.			}1	1				
Cat. I Equipment						1		
Cat. I Piping					}1	1		

*Control room readout.
#May be omitted if soil-structure interaction is negligible.
}Denotes either of the two locations.

TABLE 3.7.4-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelerographs	M	R	SA
2. Triaxial Peak Accelerographs	NA	R	NA
3. Triaxial Seismic Switches	M	R	SA
4. Triaxial Response-Spectrum Recorders	M	R	SA

Legend:

M = Monthly
R = Refueling
SA = Once per 18 months
NA = Not Applicable

NUREG-0800
(Formerly NUREG-75/087)

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3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

The MEB reviews information in the SAR concerning methods of analysis for seismic Category I components and supports, including both those designated as Code* Class 1, 2, 3, or CS and those not covered by the Code. Certain aspects of dynamic system analysis methods are discussed in Standard Review Plan Section 3.9.2 as well as this SRP section. Information is also reviewed concerning design transients for Code Class 1 and CS components and supports. The following specific subjects are reviewed under this SRP section:

1. Transients which are used in the design and fatigue analyses of all Code Class 1 and CS components, and supports and reactor internals. The Reactor Systems Branch confirms on request the acceptability of the listed transients and the number of cycles and events expected over the service lifetime of the plant. The Structural Engineering Branch confirms the seismic cyclic ground input loading as described in SRP Section 3.7.3. The method used to determine the seismic cyclic loading used for fatigue analysis of appropriate components and supports will be reviewed.

2. Description and verification of all computer programs which will be used in analyses of seismic Category I Code and non-Code items listed in this SRP section.

3. Description of any experimental stress analysis programs which will be used in lieu of theoretical stress analyses.

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*American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter "the Code").

Enclosure 3

4. Description of the analysis methods which will be used if the applicant elects to use elastic-plastic stress analysis methods in the design of · any of the above-noted components.

II. ACCEPTANCE CRITERIA

MEB acceptance criteria is based on meeting the relevant requirements of the following regulations:

1. General Design Criterion 1 as it relates to components important to safety being designed, fabricated, erected, constructed, tested and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety-function to be performed.

2. General Design Criterion 2 as it relates to safety-related mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety function.

3. General Design Criterion 14 as it relates to the reactor coolant pressure boundary being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

4. General Design Criterion 15 as it relates to the mechanical components of the reactor coolant system being designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

5. 10 CFR Part 50, Appendix B as it relates to design quality control.

6. 10 CFR Part 100, Appendix A as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

Specific criteria necessary to meet the relevant requirements of the regulations listed above are as follows:

1. To meet the requirements of GDC 1, 2, 14, 15, and 10 CFR Part 100, Appendix A, the applicant shall provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and CS components, supports and reactor internals within the reactor coolant pressure boundary. The number of events for each transient and the number of load and stress cycles per event and for events in combination shall be included. All transients such as startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, seismic events as determined from the criteria specified in Appendix A of 10 CFR Part 100, and design basis events, that are contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary shall be specified, including reactor internals and core support structures.

   The section of the applicant's SAR which pertains to transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions resulting from those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in References 8 and 9. Transients and resulting loads and load combinations with appropriate specified design and service limits must provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

2. To meet the requirements of 10 CFR Part 50, Appendix B and GDC 1, a list of computer programs that will be used (preferably programs which are recognized and widely known) in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items, and the analyses to determine stresses shall be provided. For each program the following information shall be provided to demonstrate its applicability and validity:

   a. The author, source, dated version and facility.

   b. A description, and the extent and limitation of its application.

   c. The computer program solutions to a series of test problems which shall be demonstrated to be substantially similar to solutions obtained from any one of sources 1 through 4, and source 5:

      (1) hand calculations
      (2) analytical results published in the literature
      (3) acceptable experimental tests
      (4) by an MEB acceptable similar program
      (5) the benchmark problems prescribed in Reference 10.

   A summary comparison of the solution obtained by using sources 1 through 4 shall be provided, in either graphical or numerical form. For source 5, the complete computer printout of the input and the solution shall be submitted for every benchmark problem. These solutions may be referenced, and need not be resubmitted, in subsequent license application provided the information submitted under a. and b. remains unchanged.

3. To meet the requirements of GDC 1, 14, and 15, if experimental stress analysis methods are used in lieu of analytical methods, for any seismic Category I Code or non-Code items, the section of the SAR discussing the experimental stress analysis methods will be acceptable if the information provided meets the provisions of Appendix II of Reference 7, and as in the case of analytical methods, if the information provided is sufficiently detailed to show the validity of the design to meet the provisions of the Code-required "Design Specifications."

4. To meet the requirements of GDC 1, 14, and 15 when Service Level D limits are specified by the applicant for Code Class 1 and CS components, and for supports, reactor internals, and other non-Code items, the methods of analysis used to calculate the stresses and deformations shall conform to the methods outlined in Appendix F of Reference 7, subject to the conditions discussed in subsection III.4 below.

III. REVIEW PROCEDURES

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

1. The list of transients, the number of events estimated for each transient presented in the applicant's SAR, and the method used to determine this number are compared to the same information on similar and previously licensed applications and to the acceptance criteria outlined in subsection II above. Any deviations from previous accepted practice are noted and the applicant is required to justify these deviations. For Code Class 1 and CS components and supports the MEB verifies that for each transient loading condition or combination an acceptable Code ervice limit has been specified, i.e., Design, Level A, Level B, Level C, or Level D as specified in Reference 7.

   Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

2. The information pertaining to computer programs which is presented in the applicant's SAR is reviewed as follows:

   a. The list of programs is evaluated to determine that the applicant has adequately described each program with respect to the type of analysis that is performed and the specific components to which the program is applied.

   b. The submitted computer solutions to the test problems required in subsection II.2 of this SRP section are reviewed and compared to the test solutions. Satisfactory agreement of computer and test solutions, usually within a $\pm 5\%$ error band, provides verification of the quality and adequacy of the computer programs to perform the functions for which they were designed.

   Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

3. If the applicant elects to use experimental stress analysis techniques in lieu of theoretical stress analyses, sufficient information must be presented in the SAR to demonstrate that the requirements of Appendix II to Reference 7 as they apply to the conditions set forth in the "Design Specifications" have been met.

4. If the applicant employs an elastic or an elastic-plastic method of analysis to evaluate the design of safety-related Code or non-Code items for which Service Level D limits have been specified (NB-3225 and Appendix F of Reference 7), the review covers the following points:

   a. The applicant must demonstrate that the stress-strain relationship for component materials that will be used in the analysis is valid. The ultimate strength values at service temperature must be justified.

b. The analytical procedures to be used in the analysis are reviewed to determine the validity of the analysis. If a computer program is used, the applicable requirements of subsection II.2 above shall be met.

c. If elastic system analysis is used, its application may require detailed review and justification if applied to the analysis of systems which contain active components with close tolerances, or systems in which the sequence of load application could significantly affect the actual stress distribution.

d. If elastic, elastic-plastic or limit analysis methods are used for components in conjunction with elastic or elastic-plastic system analyses, the basis upon which these procedures are used are reviewed. The applicant shall provide assurance that the calculated item or item support deformations and displacements do not violate the corresponding limits and assumptions on which the methods used for the system analysis are based.

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

IV. EVALUATION FINDINGS

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

The staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components is acceptable and meets the relevant requirements of General Design Criteria 1, 2, 14, 15, 10 CFR Part 50, Appendix B, and 10 CFR Part 100, Appendix A. This conclusion is based on the following:

1. The applicant has met the relevant requirements of General Design Criteria 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specified design and service limits which the applicant has used for designing Code Class 1 and CS components and supports, and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

2. The applicant has met the relevant requirements of General Design Criteria 2 and 10 CFR Part 100, Appendix A by including seismic events in design transients which serve as design basis to withstand the effects of natural phenomena.

3. The applicant has met the relevant requirements of 10 CFR Part 50, Appendix B, and General Design Criteria 1 by having submitted information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I Code Class 1, 2, 3, and CS structures, and non-Code structures within the present state-of-the-art limits and by having design control measures which are acceptable to assure the quality of the computer programs.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants regarding the NRC staff's plan for using this SRP section.

Except in those cases in which the applicant proposes acceptable alternative methods for complying with specified portions of the Commission's regulations, the methods described here will be used by the staff in its evaluation of comformance with Commission regulations.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, Criterion 1, "Quality Standards and Reports."

2. 10 CFR Part 50, Appendix A, Criterion 2, "Design Bases for Protection Against National Phenomena."

3. 10 CFR Part 50, Appendix A, Criterion 14, "Reactor Coolant Pressure Boundary."

4. 10 CFR Part 50, Appendix A, Criterion 15, "Reactor Coolant System Design."

5. 10 CFR Part 50, Appendix B, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants."

6. 10 CFR Part 100, Appendix A, "Reactor Site Criterion."

7. ASME Boiler and Pressure Vessel Code, Section III, Divison I, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

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

9. Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, 3 Components, Component Supports, and Core Support Structures."

10. Report NUREG/CR-1677, "Piping Benchmark Problems."

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3.9.2 DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

MEB reviews the criteria, testing procedures, and dynamic analyses employed to assure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those due to fluid flow and postulated seismic events to assure conformance with General Design Criteria 1, 2, 4, 14 and 15. The staff review covers the following specific areas:

1. Piping vibration, thermal expansion, and dynamic effect testing should be conducted during startup testing. The systems to be monitored should include (a) all ASME Code Class 1, 2, and 3 systems, (b) other high-energy piping systems inside Seismic Category I Structures, (c) high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable safety level, and (d) Seismic Category I portions of moderate-energy piping systems located outside containment. The supports and restraints necessary for operation during the life of the plant are considered to be parts of the piping system. The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been adequately designed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions anticipated during service and to confirm that normal thermal motion is not restrained. The test program description should include a list of different flow modes, a list of selected locations for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessive vibration or indications of normal thermal motion restraint occurs.

2. The following areas related to the seismic system analysis described in the applicant's safety analysis report (SAR) are reviewed.

Enclosure 4

a. Seismic Analysis Method

For all Category I systems, components, equipment and their supports (including supports for conduit and cable trays, and ventilation ducts), the applicable seismic analysis methods (response spectra, time history, equivalent static load) are reviewed. The manner in which the dynamic system analysis method is performed is reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees or freedom is reviewed. The manner in which consideration is given in the seismic dynamic analysis to maximum relative displacements between supports is reviewed. In addition, other significant effects that are accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response are reviewed.

b. Determination of Number of Earthquake Cycles

Criteria or procedures used to establish the number of earthquake cycles during one seismic event and the maximum number of cycles for which applicable Category I systems and components are designed are specified by Structural Engineering Branch (SEB) in SRP Section 3.7.3, subsection I.2.

c. Basis for Selection of Frequencies

As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed.

d. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of systems, and components are reviewed.

e. Combination of Modal Responses

When a response spectrum approach is used for calculating the seismic response of systems, or components, the phase relationship between various modes is lost. Only the maximum responses for each mode can be determined. The maximum responses for modes do not in general occur at the same time and these responses have to be combined according to some procedure selected to approximate or bound the response of the system. When a response spectra method is used, the description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) is reviewed, including that for modes with closely spaced frequencies.

f. Analytical Procedures for Piping Systems

The analytical procedures applicable to seismic analysis of piping systems, including methods used to consider differential piping support movements at different support points located within a structure and between structures, are reviewed.

g. Multiply-Supported Equipment and Components with Distinct Inputs

The criteria and procedures for seimic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are reviewed.

h. Use of Constant Vertical Static Factors

Where applicable, justification for the use of constant static factors as vertical response loads for designing Category I systems, components, equipment and their supports in lieu of the use of a vertical seismic system dynamic analysis is reviewed.

i. Torsional Effects of Eccentric Masses

The criteria and procedures that are used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic system analyses are reviewed.

j. Category I Buried Piping Systems

For Category I buried piping, the seismic criteria and methods which consider the effect of fill settlement including pipe profile and pipe stresses, the movements at support points, penetrations, and anchors are reviewed.

k. Interaction of Other Piping With Category I Piping

The seismic analysis procedures to account for the seismic motion of non-Category I piping systems in the seismic design of Category I piping are reviewed.

l. Criteria Used for Damping

The criteria to account for damping in systems, components, equipment and their supports is reviewed.

3. Dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions should be analyzed for prototype (first of a design) reactors. Generally, this analysis is not required for nonprototypes except that segments of an analysis may be necessary if there are substantial deviations from the prototype internals design. The purpose of this analysis is to predict the vibration behavior of the components, so that the input forcing functions and the level of response can be estimated. Before conducting the analyses, the specific locations for calculated responses, the considerations in defining the mathematical models, the interpretation of analytical results, the acceptance criteria, and the methods of verifying predictions by means of tests should be determined. If the reactor internal structures are a nonprototype design, reference should be made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given.

4. Flow-induced vibration testing of reactor internals should be conducted during the preoperational and startup test program. The purpose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated flow-induced

vibrations of significant magnitude or structural damage. The test program description should include a list of flow modes, a list of sensor types and locations, a description of test procedures and methods to be used to process and interpret the measured data, a description of the visual inspections to be made, and a comparison of the test results with the analytical predictions. If the reactor internal structures are a nonprototype design, reference should be made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given.

5. Dynamic system analyses should be performed to confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a loss-of-coolant accident (LOCA) in combination with the SSE. The staff review covers the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing functions, the calculational scheme, the acceptance criteria, and the interpretation of analytical results.

6. A discussion should be provided which describes the methods to be used to correlate results from the reactor internals vibration test with the analytical results from dynamic analyses of the reactor internals under steady-state and operational flow transient conditions.

   In addition, test results from previous plants of similar characteristics may be used to verify the mathematical models used for the loading condition of LOCA in combination with the SSE by comparing such dynamic characteristics as the natural frequencies. The staff review covers the methods to be used for comparison of test and analytical results and for verification of the analytical models.

Computer programs used in the analyses discussed in this SRP section are reviewed in accordance with SRP Section 3.9.1.

The Reactor Systems Branch (RSB) verifies on request that (1) the various flow modes to be used to conduct the vibration test of the reactor internals are representative of the steady-state and operational transient conditions anticipated for the reactor during its service, and (2) that an acceptable hydraulic analysis has been used to determine the loads acting on the reactor coolant system piping and the reactor internals.

II. ACCEPTANCE CRITERIA

MEB acceptance criteria are based on meeting the relevant requirements set forth in General Design Criteria 1, 2, 4, 14 and 15. The relevant requirements are as follows:

A. General Design Criterion 1, as it relates to the testing and analysis of systems, components, and equipment with appropriate safety functions being performed to appropriate quality standards.

B. General Design Criterion 2, as it relates to systems, components, and equipment important to safety being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena (SSE).

C. General Design Criterion 4, as it relates to systems and components important to safety being appropriately protected against the dynamic effects of discharging fluids.

D. General Design Criterion 14, as it relates to systems and components of the reactor coolant pressure boundary being designed so as to have an extremely low probability of rapidly propagating failure or of gross rupture.

E. General Design Criterion 15, as it relates to the reactor coolant system being designed with sufficient margin to assure that the reactor coolant pressure boundary will not be breeched during normal operating conditions including anticipated operational occurrences.

Specific criteria necessary to meet the relevant requirements of the Commission regulations identified above are as follows:

1. Relevant requirements of GDC 14 and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during startup functional testing for specified high-and moderate-energy piping, and their supports and restraints. The purpose of these tests is to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service as required by the Code and to confirm that no unacceptable restraint of normal thermal motion occurs.

   An acceptable test program to confirm the adequacy of the designs should consist of the following:

   a. A list of systems that will be monitored.

   b. A listing of the different flow modes of operation and transients such as pump trips, valve closures, etc. to which the components will be subjected during the test. (For additional guidance see Reference 8.) For example, the transients associated with the reactor coolant system heatup tests should include, but not necessarily be limited to:

      (1) Reactor coolant pump start.

      (2) Reactor coolant pump trip.

      (3) Operation of pressure-relieving valves.

      (4) Closure of a turbine stop valve.

   c. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak) or other appropriate criteria, to be used to show that the stress and fatigue limits are within the design levels, should be provided.

   d. A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position.

e. A description of the thermal motion monitoring program, i.e., verification of snubber movement, adequate clearances and gaps, including acceptance criteria and how motion will be measured.

f. If vibration is noted beyond the acceptance levels set by the criteria of c., above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in d., above, a description should be provided of the corrective action to be taken to assure that the snubber is operable.

2. To meet the relevant requirements of GDC 2, the acceptance criteria for the areas of review described in subsection I.2 of this SRP section are given below. Other approaches which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used to confirm the ability of all seismic Category I systems, components, equipment, and their supports to function as needed during and after an earthquake.

a. Seismic Analysis Methods

The seismic analysis of all Category I systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts) should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified.

(1) Dynamic Analysis Method

A dynamic analysis (e.g., response spectrum method, time history method, etc.) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items:

(a) Use of either the time history method or the response spectrum method.

(b) Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all Category I and applicable non-Category I systems and plant equipment. The number is considered adequate when additional degrees of freedom do not result in more than a 10% increase in responses. Alternately, the number of degrees of freedom may be taken equal to twice the number of modes with frequencies less than 33 hz.

(c) Investigation of a sufficient number of modes to assure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10% increase in responses.

(d) Consideration of maximum relative displacements among supports of Category I systems, and components.

(e) Inclusion of significant effects such as piping interactions, externally applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses.

(2) Equilvalent Static Load Method

An equilvalent static load method is acceptable if:

(a) Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Typical examples or published results for similar systems may be submitted in support of the use of the simplified method.

(b) The design and associated simplified analysis account for the relative motion between all points of support.

(c) To obtain an equivalent static load of equipment or component which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used if adequate justification is provided.

In addition, for equipment which can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable. For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable.

b. Determination of Number of Earthquake Cycles

During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) should be assumed. The number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed (extract from SRP Section 3.7.3, subsection II.2).

c. Basis for Selection of Frequencies

To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.

d. Three Components of Earthquake Motion

Depending upon what basic methods are used in the seismic analysis, i.e., response spectra or time history method, the following two approaches are considered acceptable for the combination of three-dimensional earthquake effects. (Ref. 11, 12, and 13)

(1) Response Spectra Method

When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model.

(2) Time History Analysis Method

When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. (a) to obtain maximum responses due to each of the three components of the earthquake motion: in this case the method for combining the three-dimensional effects is identical to that described in (a) except that the maximum responses are calculated using the time history method instead of the spectrum method. (b) To obtain time history responses from each of the three components of the earthquake motion and combine them at each time step algebraically: the maximum response in this case can be obtained from the combined time solution. When this method is used, to be acceptable, the earthquake motions specified in the three different directions should be statistically independent.

e. Combination of Modal Responses

When the response spectrum method of analysis is used to determine the dynamic response of damped linear systems, the most probable response is obtained as the square root of the sum of the squares of the responses from individual modes. Thus, the most probable system response, R, is given by

$$R = \left(\sum_{k=1}^{N} R_k^2\right)^{1/2} \qquad (1)$$

where R_k is the response for the k^{th} mode and N is the number of significant modes considered in the modal response combination.

When modes with closely spaced modal frequencies exist, an acceptable method for obtaining the system response is to take the absolute sum of the responses of the closely spaced modes and combine this sum with other remaining modal responses using the square root of the sum of the squares rule. Two modes having frequencies within 10% of each other are considered as modes with closely spaced frequencies.

This approach is simple and straightforward in all those cases where the group of modes with closely spaced frequencies is tightly bundled, i.e., the lowest and the highest modes of the group are within 10% of each other. However, when the group of closely spaced modes is spaced widely over the frequency range of interest (while the frequencies of the adjacent modes are closely spaced), the absolute sum method of combining responses tends to yield over-conservative results. To obviate this problem, a general approach applicable to all modes is considered appropriate. The following equation is merely a mathematical representation of this approach.

The most probable system response, R, is given by

$$R = \left(\sum_{k=1}^{N} R_k^2 + 2 \sum \left| R_\ell R_m \right| \right)^{1/2} \qquad (2)$$

Where the second summation is to be done on all ℓ and m modes whose frequencies are closely spaced to each other.

Other approaches which give an equivalent degree of conservatism to the above methods, and which are adequately justified are also acceptable. Regulatory Guide 1.92 (Reference 10) "Combining Modal Responses and Spatial Components in Seismic Response Analysis" presents detailed guidance on this topic.

f. Analytical Procedures for Piping Systems

The seismic analysis of Category I piping may use either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are as given in subsection II.2.a of this SRP section.

g. Multiply-Supported Equipment and Components With Distinct Inputs

Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, by using the floor response spectra. For the latter option, the maximum displacement of each support is predicted by $S_d = S_a g/\omega^2$, where S_a is the spectral acceleration in "g's" at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method.

In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra involves determination of dynamic responses due to the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors and applied identically to all the floors, provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination using static analysis procedures.

In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the systems (Ref. 16). Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiply-supported systems.

h. Use of Constant Vertical Static Factors

The use of constant vertical load factors as vertical response loads for the seismic design of all Category I systems, components, equipment, and their supports in lieu of the use of a vertical seismic system dynamic analysis is acceptable only if it can be justified that the structure is rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction is more than 33 hz.

i. Torsional Effects of Eccentric Masses

For seismic Category I systems, if the torsional effect of an eccentric mass such as a valve operator in a piping system is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical mode. The criteria for significance will have to be determined on a case-by-case basis.

j. Category I Buried Piping Systems

For Category I buried piping systems, the following items should be considered in the analysis:

(1) The inertial effects due to an earthquake upon buried piping systems should be adequately accounted for in the analysis. Use of the procedures described in References 11 and 14 is acceptable.

(2) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Use of the procedures described in Reference 15 is acceptable.

(3) When applicable, the effects due to local soil settlements, soil arching, etc., should also be considered in the analysis.

k. Interaction of Other Piping with Category I Piping

To be acceptable, each non-Category I piping system should be designed to be isolated from any Category I piping system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I piping system. If it is not feasible or practical to isolate the Category I piping system, adjacent non-Category I piping should be analyzed according to the same seismic criteria as applicable to the Category I piping system. For non-Category I piping systems attached to Category I piping systems, the dynamic effects of the non-Category I piping should be simulated in the modeling of the Category I piping. The attached non-Category I piping, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I piping.

l. Criteria Used for Damping

Regulatory Guide 1.61 (Reference 9) "Damping Values for Seismic Design of Nuclear Power Plants," provides acceptable values which may be used. The use of alternate damping values requires justification.

3. Relevant requirements of GDC 1 and 4 are met as given below. The following guidelines, in addition to Regulatory Guide 1.20 (Reference 7), apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs.

a. The results of vibration calculations for a prototype reactor should consist of the following:

(1) Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, and the response causing the maximum stress amplitude should be calculated.

(2) The dynamic properties of internal structures, including the natural frequencies, the dominant mode shapes, and the damping factors should be characterized. If analyses are performed on a component structural element basis, the existence of dynamic coupling among component structure elements should be investigated.

(3) The response characteristics, such as the dependence on hydrodynamic excitation forces, the flow path configuration, coolant recirculation pump frequencies, and the natural frequencies of the internal structures, should be identified.

(4) Acceptance criteria for allowable responses should be established, as should criteria for the location of vibration sensors. Such criteria should be related to the Code allowable stresses, strains, and limits of deflection that are established to preclude loss of function with respect to the reactor core structures and fuel assemblies.

b. The forcing functions should account for the effects of transient flow conditions and the frequency content. Acceptable methods for formulating forcing functions for vibration prediction include the following:

(1) Analytical method: based on standard hydrodynamic theory, the governing differential equations for vibratory motions should be developed and solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable.

(2) Test-analysis combination method: based on data obtained from plant tests or scaled model tests (e.g., velocity or pressure distribution data), forcing functions should be formulated which will include the effects of complex flow path configurations and wide variations of pressure distributions.

(3) Response-deduction method: based on a derivation of response characteristics from plant or scaled model test data, forcing functions should be formulated. However, since such functions may not be unique, the computational procedures and the basis for the selection of the representative forcing functions should be described.

c. Acceptable methods of obtaining dynamic responses for vibration predictions are as follows:

(1) Force-response computations are acceptable if the characteristics of the forcing functions are predetermined on a conservative basis and the mathematical model of the reactor internals is appropriately representative of the design.

(2) If the forcing functions are not predetermined, either a special analysis of the response signals measured from reactor internals of similar design may be performed to predict amplitude and modal contributions, or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used.

d. Vibration predictions should be verified by test results. If the test results differ substantially from the predicted response behavior, the vibration analysis should be appropriately modified to improve the agreement with test results and to validate the analytical method as appropriate for predicting responses of the prototype unit, as well as of other units where confirmatory tests are to be conducted.

4. Relevant requirements of GDC 1 and 4 are met as given below. The preoperational vibration test program for the internals of a prototype (first of a design) reactor should conform to the requirements for a prototype test, as specified in Regulatory Guide 1.20, including vibration prediction, vibration monitoring, data reduction, and surface inspection. The test program to demonstrate design adequacy of the reactor internals should include, but not necessarily be limited to the following:

   a. The vibration testing should be conducted with the fuel elements in the core or with dummy elements which provide equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if it can be demonstrated that testing in this mode is conservative.

   b. A brief description of the vibration monitoring instrumentation should be provided, including instrument types and diagrams of locations, which should include the locations having the most severe vibratory motions or having the most effect on safety functions.

   c. The planned duration of the test for the normal operation modes to assure that all critical components are subjected to at least 10^6 cycles of vibration should be provided. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 1.2 days or more will be acceptable.

   d. Testing should include all of the different flow modes of normal operation and upset transients. The proposed set of flow modes are acceptable if they provide a conservative basis for determining the dynamic response of the reactor internals and are reviewed by RSB on request.

   e. The methods and procedures to be used to process the test data to obtain a meaningful interpretation of the core structure vibration behavior should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and the possible effects on safety functions.

   f. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test.

   g. Visual and nondestructive surface inspections should be performed after the completion of the vibration tests. The inspection program description should include the areas subject to inspection, the methods of inspection, the design access provisions to the reactor internals, and the equipment to be used for performing such inspections. These inspections should be conducted preferably following the removal of the internals from the reactor vessel. Where removal is not feasible, the inspections should be performed by means of equipment appropriate for in situ inspection. The areas inspected should include all load-bearing interfaces, core restraint devices, high stress locations, and locations critical to safety functions.

For internals of subsequent reactors that have the same design, size, configuration, and operating conditions as the prototype reactor internals, the vibration test program should conform to the requirements of the appropriate nonprototype program as specified in Regulatory Guide 1.20.

5. Relevant requirements of GDC 2 and 4 are met as given below. Dynamic system analyses should be performed to confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically.

   The most severe dynamic effects from LOCA loadings are generally found to result from a postulated double-ended rupture of a primary coolant loop near a reactor vessel inlet or outlet nozzle with the reactor in the most critical normal operating mode. However, all other postulated break locations should be evaluated and the location producing the controlling effects should be identified.

   Mathematical models used for dynamic system analysis for LOCA in combination with the SSE effects should include the following:

   a. Modeling should include reactor internals and dynamically related piping, pipe supports, components, and fluid-structure interaction effects when applicable. Typical diagrams and the basis of modeling should be developed and described.

   b. Mathematical models should be representative of system structural characteristics, such as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).

   c. Any system structural partitioning and directional decoupling employed in the dynamic system modeling should be justified.

   d. The effects of flow upon the mass and flexibility properties of the system should be discussed.

   Typical diagrams and the basis for postulating the LOCA-induced forcing function should be provided, including a description of the governing hydrodynamic equations and the assumptions used for mathematically tractable flow path geometries, tests for determining flow coefficients, and any semiempirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions. The acceptabilty of the hydraulic analysis, as reviewed by RSB on request, is based on established engineering practice and generic topical reviews performed by the staff.

The methods and procedures used for dynamic system analyses should be described, including the governing equations of motion and the computational scheme used to derive results. Time domain forced-response computation is acceptable for both LOCA and SSE analyses. The response spectrum modal analysis method may be used for SSE analysis.

The stability of elements in compression, such as the core barrel and the control rod guide tubes under outlet pipe rupture loadings should be investigated.

Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports.

The criteria for acceptance of the analytical results are as provided in SRP Sections 3.9.3 and 3.9.5.

6. Relevant requirements of GDC 1 are met as given below. Regarding the correlation to be made of tests and analyses of reactor internals, a discussion covering the following items to assure the adequacy and sufficiency of the test and analysis results should be provided:

   a. Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for possible verification of the mathematical model used in the analysis.

   b. Comparison of the analytically obtained mode shapes with the shape of measured motion for possible identification of the modal combination or verification of a specific mode.

   c. Comparison of the response amplitude time variation and the frequency content obtained from test and analysis for possible verification of the postulated forcing function.

   d. Comparison of the maximum responses obtained from test and analysis for possible verification of stress levels.

   e. Comparison of the mathematical model used for dynamic system analysis under operational flow transients and under the combined LOCA and SSE loadings, to note similarities.

III. REVIEW PROCEDURES

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

General Design Criteria 1, 2, 4, 14, and 15 state that all structures, system and components important to safety should be designed and tested to assure that safety functions can be performed in the event of operational transients, earthquakes, and LOCA loadings.

Under these guidelines, the staff reviews the treatment of dynamic responses of safety-related piping systems and reactor internal structures by the following procedures:

1. During the CP stage, the PSAR is reviewed to assure that the applicant has provided a commitment to conduct a piping steady-state vibration, thermal expansion and operational transient test program. The applicants program description should be sufficiently comprehensive to contain all the elements of an acceptable program as described in subsection II.1 of this SRP Section.

   During the OL stage, the FSAR is reviewed to assure that the applicant's PSAR commitment is fulfilled and the program is developed in sufficient detail. The reviewer should be assured that the applicants program as described in Sections 3.9.2 and 14.0 of the FSAR is sufficiently developed to:

   a. Establish the rationale and bases for the acceptance criteria and selection of locations to monitor pipe motions.

   b. Provide the displacement or other appropriate limits at locations to be monitored.

   c. Describe the techniques and instruments (as needed) for monitoring or measuring pipe motions.

   d. Assure that the NRC will be provided documentation of any corrective action resulting from the test and conformation by additional testing that substantiates effectiveness of the corrective action.

2. For seismic system analysis review, the following review procedures are implemented.

   a. Seismic Analysis Methods

      For all Category I systems, components, equipment and their supports (including supports for conduit and cable trays, and ventilation ducts), the applicable methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to ascertain that the techniques employed are in accordance with the acceptance criteria as given in subsection II.2.a of this SRP section.

   b. Determination of Number of Earthquake Cycles

      Criteria or procedures used to establish the number of earthquake cycles are reviewed to determine that they are in accordance with the acceptance criteria as given in subsection II.2.b of this SRP section. Justification for deviating from the acceptance criteria is requested from the applicant, as necessary.

   c. Basis for Selection of Frequencies

      As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed to determine compliance with the acceptance criteria of subsection II.2.c of this SRP section.

   d. Three Components of Earthquake Motion

      The procedures by which the three components of earthquake motion are considered in determining the seismic response of systems are reviewed to determine compliance with the acceptance criteria of subsection II.2.d of this SRP section.

e. Combination of Modal Responses

The procedures for combining modal responses are reviewed to determine compliance with the acceptance criteria of subsection II.2.e of this SRP section, when a response spectrum modal analysis method is used.

f. Analytical Procedures for Piping Systems

For all Category I piping and applicable non-Category I piping, the methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to determine that the techniques employed are in accordance with the acceptance criteria of subsection II.2.f of this SRP section. Typical mathematical models are reviewed to judge whether all significant degrees of freedom have been included.

g. Multiply-Supported Equipment and Components With Distinct Inputs

The criteria for the seismic analysis of multiply-supported components and equipment with distinct inputs are reviewed to determine that the criteria are in accordance with the acceptance criteria of subsection II.2.g of this SRP section.

h. Use of Constant Vertical Static Factors

Use of constant static factors as response loads in the vertical direction for the seismic design of any Category I systems in lieu of a detailed dynamic method is reviewed to determine that constant static factors are used only if the structure is rigid in the vertical direction based on the definition for rigidity given in subsection II.2.h of this SRP section.

i. Torsional Effects of Eccentric Masses

The procedures for seismic analysis of Category I piping systems are reviewed to determine compliance with the acceptance criteria of subsection II.2.i of this SRP section.

j. Category I Buried Piping Systems

The analysis procedures for Category I buried piping are reviewed to determine that they are in accordance with the acceptance criteria of subsection II.2.j of this SRP section. This includes review of the procedures used to consider the effect of fill settlement including pipe profile and pipe stresses, and the differential movements at support points, penetrations, and anchors. Any procedures that are not adequately justified are so identified and the applicant is requested to provide additional justification.

k. Interaction of Other Piping with Category I Piping

The criteria used to design the interfaces between Category I and non-Category I piping are reviewed to determine compliance with the acceptance criteria of subsection II.2.k of this SRP section.

l. Criteria used for Damping

The criteria used to account for damping in systems, components, equipment and their supports is reviewed to determine that it is in accordance with the regulatory position in Reference 9.

3. At the CP stage, the applicant should commit to performing an analysis of the vibration of the reactor internal structures if they are designated as a prototype design. A brief description of the methods and procedures to be used for the analysis should be provided.

At the OL stage, a detailed dynamic analysis should be provided for a prototype design, to be used for vibration prediction prior to the performance of preoperational vibration tests. Acceptance of the analysis is based on the technical soundness of the analytical method and procedures used and the degree of conformance to the acceptance criteria listed above. In addition, the analysis is verified by correlation with the test results when these are available.

For both CP and OL stages, if the reactor internal structures are a non-prototype design, then reference should be made to the reactor which is prototypical of the reactor being reviewed. A brief summary of test and analysis results for the prototype should be given. Alternatively, the information may be contained in another applicable document, such as a topical report, to which reference should be made.

4. At the CP stage, the staff review of the program for preoperational vibration testing of reactor internals for flow-induced vibrations includes the following matters:

a. The applicant should clarify his intention to perform either a prototype test or a non-prototype test.

b. If the plant is designated as a prototype, a brief description of the preoperational vibration test program should be provided. The staff review will be based on the conformance of this program to the requirements as listed in subsection II.4, above.

c. If the plant is a non-prototype, the applicant should identify the existing plant of similar design that is the prototype plant. The staff reviews the validity of the designated prototype, including any design difference of reactor internal structures from the prototype plant to verify that any design modifications do not substantially alter the behavior of the flow transients and the response of the reactor internals. Additional detailed analysis, scaled model tests, or installation of some instrumentation during the confirmatory test may be required in order to complete the review. In addition, the applicant should commit to performing the prototype test if adequate test results are not obtained on a timely basis for the designated prototype.

At the OL stage, the staff review includes the following procedures:

(1) A detailed preoperational vibration test program and the tentative schedule to perform the test are reviewed. If elements of the program differ substantially from the guidelines specified in Regulatory Guide 1.20, discussion of the need and justification for the differences should be given. On request, RSB verifies that the flow modes to be used are acceptable.

(2) For a prototype plant, the review covers the acceptability of vibration prediction, the visual surface inspection procedures, the details of instrumentation for vibration monitoring, the methods and procedures to process the test results, and possible supplementary tests, such as component vibration tests, flow tests, and scaled model tests.

(3) For a nonprototype plant, the staff verifies the applicability of the designated prototype, including the design similarity of the reactor internal structures to the prototype. Additional detailed analysis, scaled model tests, or vibration monitoring in the confirmatory tests may be needed in order to complete the review.

5. In the CP stage review of the dynamic analysis of the reactor internals and unbroken loops of the reactor coolant piping under faulted condition loadings, the applicant commits to perform this analysis or identifies the applicable document, generally in the form of a topical report, containing the required information. A brief description of the scope and methods of analysis should be provided.

   In the OL review, the staff reviews the detailed information to confirm that an adequate analysis has been made of the capability of reactor internal structures and unbroken loops to withstand dynamic loads from the most severe LOCA in combination with the safe shutdown earthquake. The staff review covers the analytical methods and procedures, the basis of the forcing functions, the mathematical models to represent the dynamic system, and the stability investigations for the core barrel and essential compressive elements. Acceptance of the analysis is based on (1) the technical soundness of the analytical methods used, (2) the degree of conformance to the acceptance criteria listed above, and (3) verification that stresses under the combined loads are within allowable limits of the applicable code and deformations are within the limits set to assure the ability of reactor internal structures and piping to perform needed safety functions. On request, RSB verifies that an acceptable hydraulic analysis has been used.

6. MEB reviews the program which the applicant has committed to implement as part of the preoperational test procedure, principally to correlate the test measurements with the analytically predicted flow-induced dynamic response of the reactor internals. MEB reviews the applicant's statements in this area to assure that there is a commitment to submit a report on a timely basis. The report should summarize the analyses and test results so that MEB can review the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

IV. EVALUATION FINDINGS

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

The staff concludes that the dynamic testing and analysis of systems, components, and equipment is acceptable and meets the relevant requirements of General Design Criteria 1, 2, 4, 14 and 15. This conclusion is based on the following:

1. The applicant has met the relevant requirements of General Design Criteria 14 and 15 with respect to the design and testing of the reactor coolant pressure boundary to assure that there is a low probability of rapidly propagating failure and of gross rupture and to assure that design conditions are not exceeded during normal operation including anticipated operational occurrences by having an acceptable vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high- and moderate-energy piping, and all associated systems, restraints and supports. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation.

2. The applicant has met the relevant requirements of General Design Criteria 2 with respect to demonstrating design adequacy of all Category I systems, components, equipment and their supports to withstand earthquakes by meeting the regulatory positions of Regulatory Guides 1.61 and 1.92 and by providing an acceptable seismic systems analysis procedure and criteria. The scope of review of the seismic system analysis included the seismic analysis methods of all Category I systems, components, equipment and their supports. It included review of procedures for modeling, inclusion of torsional effects, seismic analysis of Category I piping systems, seismic analysis of multiply-supported equipment and components with distinct inputs, justification for the use of constant vertical static factors and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of non-Category I piping with Category I piping. The review has also included criteria and seismic analysis procedures for reactor internals and Category I buried piping outside containment.

   The system analyses are performed by the applicant on an elastic basis. Modal response spectrum multidegree of freedom and time history methods form the bases for the analyses of all major Category I systems, components, equipment and their supports. When the modal response spectrum method is used, governing response parameters are combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses are used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of systems, components, equipment and their supports are generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be employed for all systems, and components, equipment and their supports where analyses show significant structural amplification in the vertical direction.

3. The applicant has met the relevant requirements of General Design Criteria 1 and 4 with respect to the reactor internals being designed and tested to quality standard commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects by meeting the regulatory positions of Regulatory Guide 1.20 for the conduct of preoperational vibration tests and by having a preoperational vibration program planned for the reactor internals which provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown.

4. The applicant has met the relevant requirements of General Design Criteria 2 and 4 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the safe shutdown earthquake (SSE) by having a dynamic system analysis to be performed which provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of postulated loss of coolant accidents (LOCA) and the SSE and the combined loads of a postulated main steam line rupture and SSE (for a BWR). The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events.

5. The applicant has met the relevant requirements of General Design Criterion 1 with respect to systems and components being designed and tested to quality standards commensurate with the importance of the safety functions to be performed by the proposed program to correlate the test measurements with the analysis results. The program constitute an acceptable basis for demonstrating the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

   For the FSAR, the review should provide justification for a finding similar to that stated above with the phrase "will be implemented" modified to read "has been implemented."

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced Regulatory Guides.

VI. REFERENCES

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

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

5 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

6. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

7. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

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

9. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

10. Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

11. N. M. Newmark, J. A. Blume, and K. K. Kapur, "Design Response Spectra for Nuclear Power Plants," Journal of the Power Division, American Society of Civil Engineers, pp. 287-303, November 1973.

12. S. L. Chu, M. Amin, and S. Singh, "Spectral Treatment of Actions of Three Earthquake Components on Structures," Nuclear Engineering and Design, Vol. 21, pp. 126-136 (1972).

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15. M. Hetenyi, "Beams on Elastic Foundation," The University of Michigan Press (1946).

16. R. P. Kassawara, and D. A. Peck, "Dynamic Analysis of Structural Systems Excited at Multiple Support Locations," 2nd ASCE Specialty Confrerence on Structural Design of Nuclear Plant Facilities, Chicago, Dec. 17-18, 1973.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

The MEB reviews the information presented in the applicant's safety analysis report (SAR) concerning the structural integrity of pressure-retaining components, their supports, and core support structures which are designed in accordance with the rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1 (hereinafter "the Code") (Reference 3) and General Design Criteria 1, 2, 4, 14, and 15 (Reference 2).

The staff reviews covers the following specific areas:

1. Loading Combinations, System Operating Transients, and Stress Limits

The design and service loading combinations (e.g., design and service loads, including system operating transients, in combination with loads calculated to result from postulated seismic and other events) specified for Code constructed items designated as Code Class 1, 2, 3 (including Class 1, 2 and 3 component support structures) and CS core support structures are reviewed to determine that appropriate design and service limits have been designated for all loading combinations. This review ascertains that the design and service stress limits and deformation criteria comply with the applicable limits specified in the Code and Appendix A to this SRP section. Service stress limits which allow inelastic deformation of Code Class 1, 2, and 3 components, component supports, and Class CS core support structures are evaluated as are the justifications for the proposed design procedures. Piping which is "field run" should be included. Internal parts of components, such as valve discs and seats and pump shafting, subjected to dynamic loading during operation of the component should be included.

Enclosure 5

2. Design and Installation of Pressure Relief Devices

The design and installation criteria applicable to the mounting of pressure relief devices (safety valves and relief valves) for the overpressure protection of Code Class 1, 2, and 3 components are reviewed. The review includes evaluation of the applicable loading combinations and stress criteria. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system. The dynamic structural response due to BWR safety relief valve discharge into the suppression pool is also considered.

The design of safety and relief valve systems is reviewed with respect to the load combinations imposed on the safety or relief valves, upstream piping or header, downstream or vent piping, system supports, and BWR suppression pool discharge devices such as ramsheads and quenchers.

The load combinations should identify the most severe combination of the applicable loads due to internal fluid weight, momentum and pressure, dead weight of valves and piping, thermal load under heatup, steady state and transient valve operation, reaction forces when valves are discharging (thrust, bending, and torsion), seismic forces, and dynamic forces due to BWR safety relief valve discharge into the suppression pool as applicable. The reaction loads due to discharge of loop seal water slugs and subcooled or saturated liquid under transient or accident conditions shall also be included as valve discharge loads.

The structural response of the piping and support system is reviewed with particular attention to the dynamic or time history analyses employed in evaluating the appropriate support and restraint stiffness effects under dynamic loadings when valves are discharging.

Where the use of hydraulic snubbers is proposed, the snubber performance characteristics are reviewed to assure that their effects have been considered in the analyses under steady state valve operation and repetitive load applications caused by cyclic valve opening and closing during the course of a pressure transient.

3. Component Supports

The review of information submitted by the applicant includes an evaluation of Code Class 1, 2, and 3 components supports. The review includes an assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. All the component supports of these three types are covered in this SRP section. Although classified as component standard supports, snubbers require special consideration due to their unique function. Snubbers provide no load path or force transmission during normal plant operations but function as rigid supports when subjected to dynamic transient loads. Component supports are those metal supports which are designed to transmit loads from the pressure-retaining boundary of the component to the building structure. The methods of analysis for calculating the responses of the reactor coolant pressure boundary supports resulting from the combination of LOCA and SSE events are reviewed in SRP Sections 3.6.2 and 3.9.2.

In addition, the MEB will coordinate other branches evaluations that interface with the overall review of this SRP section as follows: The Equipment Qualification Branch (EQB) evaluates the operability of pumps and valves and judges the design criteria for pressure-relieving devices which may have an active function during and after a faulted plant condition against the requirements of the component operability assurance program as part of its primary review responsibility for SRP Section 3.10. The Auxiliary Systems Branch (ASB) verifies that the number and size of valves specified for the steam and feedwater systems have adequate pressure-relieving capacity as part of its primary review responsibility for SRP Section 10.3. The Reactor Systems Branch (RSB) verifies that the number and size of valves specified for the reactor coolant pressure boundary have adequate pressure-relieving capacity as part of its primary review responsibility for SRP Section 5.2.2. The Containment Systems Branch (CSB) reviews the applicant's analyses of subcompartment differential pressures resulting from postulated pipe breaks as part of its primary review responsibility for SRP Section 6.2.1.2.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

MEB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. 10 CFR Part 50, §50.55a and General Design Criterion 1 as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

B. General Design Criterion 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

C. General Design Criterion 4 as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions.

D. General Design Criterion 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

E. General Design Criterion 15 as it relates to the reactor coolant system being designed with sufficient margin to assure that the design conditions are not exceeded.

Specific criteria necessary to meet the relevant requirements of §50.55a and General Design Criteria 1, 2, 4, 14, and 15 by which the areas of review defined in subsection I of this SRP section judged to be acceptable are as follows:

1. Loading Combinations, System Operating Transients, and Stress Limits

The design and service loading combinations, including system operating transients, and the associated design and service stress limits considered for each component and its supports should be sufficiently defined to provide the basis for design of Code Class 1, 2, and 3 components, component supports and Class CS core support structures for all conditions.

The acceptability of the combination of design and service loadings (including system operating transients), applicable to the design of Class 1, 2, and 3 components, component supports, and Class CS core support structures, and of the designation of the appropriate design or service stress limit for each loading combination, is judged by comparison with positions stated in Appendix A, and with appropriate standards acceptable to the staff developed by professional societies and standards organizations.

The design criteria for internal parts of components such as valve discs, seats, and pump shafting should comply with applicable ASME Code or Code Case criteria. In those instances where no ASME criteria exist, the design criteria are acceptable if they assure the structural integrity of the part such that no safety-related functions are impaired.

2. Design and Installation of Pressure Relief Devices

The applicant should use design criteria for pressure relief stations specified in Appendix O, ASME Code, Section III, Division 1, "Rules for the Design of Safety Valve installations" (Reference 6). Additionally, the following criteria are applicable:

(1) Where more than one valve is installed on the same run pipe, the sequence of valve openings to be assumed in analyzing for the stress at any piping location should be that sequence which is estimated to induce the maximum instantaneous value of stress at that location.

(2) Stresses should be evaluated, and applicable stress limits should be satisfied for all components of the run pipe and connecting systems and the pressure relief valve station including supports and all connecting welds between these components.

(3) In meeting the stress limit requirements, the contribution from the reaction force and the moments resulting from that force should include the effects of the Dynamic Load Factor or should use the maximum instantaneous values of forces and moments for that location as determined by the dynamic hydraulic/structural system analysis. This requirement should be satisfied in demonstrating satisfaction of all design limits at all locations of the run pipe and the pressure relief valve for Class 1, 2, and 3 piping. A Dynamic Load Factor (DLF) of 2.0 may be used in lieu of a dynamic analysis to determine the DLF.

The SAR must also include a description of the calculational procedures, computer programs, and other methods to be used in the analysis. The analysis must include the time history or equivalent effects of changes of momentum due to fluid flow changes of direction. The fluid states considered must include postulated water slugs where water seals are used and subcooled or saturated liquid if such fluid can be discharged under postulated transient or accident conditions. Plants utilizing suppression pools shall also consider the applicable pool dynamic loads on the safety relief valve system. Stress computations and stress limits must be in accord with applicable rules of the Code.

3. Component Supports

a. The component support designs should provide adequate margins of safety under all combinations of loadings. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination should meet the criteria in Appendix A and Regulatory Guides 1.124 and 1.130 (References 7 and 8).

Component supports of active pumps and valves should be considered in context with the other features of the operability assurance program as presented in SRP Section 3.10. If the component support affects the operability requirements of the supported component, then deformation limits should also be specified. Such deformation limits should be compatible with the operability requirements of the components supported and incorporated into the operability assurance program. In establishing allowable deformations, the possible movements of the support base structures must be taken into account.

b. Where snubbers are utilized as supports for safety-related systems and components, acceptable criteria for snubber operability assurance should contain the following elements:

(1) Structural Analysis and Systems Evaluation.

Systems and components which utilize snubbers as shock and vibration arrestors must be analyzed to ascertain the interaction of such devices with the systems and components to which they are attached. Snubbers may be used as shock and vibration arrestors and in some instances as dual purpose snubbers. When used as a vibration arrestor or dual purpose snubbers, fatigue strength must be considered. Important factors in the fatigue evaluation include: (i) unsupported system component movement or amplitude, (ii) force imparted to snubber and corresponding reaction on system or component due to restricting motion (damped amplitude), (iii) vibration frequency or number of load cycles, and (iv) verification of system or component and snubber fatigue strength.

Snubbers used as shock arrestors do not require fatigue evaluation if it can be demonstrated that (i) the number of load cycles which the snubber will experience during normal plant operating conditions is small (<2500) or (ii) motion during normal plant operating conditions does not exceed snubber dead band.

Snubbers utilized in systems or components which may experience high thermal growth rates either during normal operating conditions or as a result of anticipated transients should be checked to assure that such thermal growth rates do not exceed the snubber lock-up velocity.

(2) Characterization of Mechanical Properties.

A most important aspect of the structural analysis is realistic characterization of snubber mechanical properties (i.e. spring rates) in the analytical model. Since the "effective" stiffness of a snubber is generally greater than that for the snubber support assembly (i.e., the snubber plus clamp, transition tube extension, back-up support structure, etc.) the snubber response characteristics may be "washed out" by the added flexibility in the support structure. The combined effective stiffness of the snubber and support assembly must therefore be considered in evaluating the structural response of the system or component.

Snubber spring rate should be determined independent of clearance/lost motion, activation level, or release rate. The stiffness should be based on structural and hydraulic compliance only, and should consider the effects of temperature.

The snubber end fitting clearance and lost motion must be minimized and should be considered when calculating snubber reaction loads and stress which are based on a linear analysis of the system or component. This is especially important in multiple snubber applications where mismatch of end fitting clearance has a greater effect on the load sharing of these snubbers than does the mismatch of activation level or release rate. Equal load sharing of multiple snubber supports should not be assumed if mismatch in end fitting clearance exists.

(3) Design Specifications

The required structural and mechanical performance of snubbers is determined from the user's system analysis described in (1) and (2). The snubber Design Specification is the instrument provided by the purchaser to the supplier to assure that the requirements are met. The Design Specification should contain (i) the general functional requirements, (ii) operating environment, (iii) applicable codes and standards, (iv) materials of construction and standards for hydraulic fluids and lubricants, (v) environmental, structural, and performance design verification tests, (vi) production unit functional verification tests and certification, (vii) packaging, shipping, handling, and storage requirements, and (viii) description of provisions for attachments and installation.

In addition, the snubber manufacturer should be requested to submit his quality assurance and assembly quality control procedures for review and acceptance by the purchaser.

(4) Installation and Operability Verification

Assurance that all snubbers and properly installed prior to preoperational piping vibration and plant start-up tests should be provided. Visual observation of piping systems and measurement of thermal movements during plant start-up tests could verify that snubbers are operable (not locked up). Provisions for such examinations and measurements should be discussed in the piping preoperational vibration and plant start-up test programs as described in SRP Section 3.9.2.

(5) Use of Additional Snubbers

Snubbers could in some instances be installed during or after plant construction which may not have been included in the design analysis. This could occur as a result of unanticipated piping vibration as discussed in SRP Section 3.9.2 or interference problems during construction. The effects of such installation should be fully evaluated and documented to demonstrate that normal plant operations and safety are not diminished.

(6) Inspection and Testing

Inservice inspection and testing are critical elements of operability assurance programs for mechanical components. The applicant should provide a discussion of accessibility provisions for maintenance, inservice inspection and testing, and possible repair or replacement of snubbers consistent with the requirements of the NRC Standard Technical Specifications.

(7) Classification and Identification

All safety-related components which utilize snubbers in their support systems should be identified and tabulated in the FSAR. The tabulation should include the following information: (i) identification of the systems and components in those systems which utilize snubbers, (ii) the number of snubbers utilized in each system and on components in that system, (iii) the type(s) of snubber (hydraulic or mechanical) and the corresponding supplier identified, (iv) specify whether the snubber was constructed to the rules of ASME Code Section III, Subsection NF, (v) state whether the snubber is used as a shock, vibration, or dual purpose snubber, and (vi) for snubbers identified as either dual purpose or vibration arrestor type, indicate if both snubber and component were evaluated for fatigue strength.

III. REVIEW PROCEDURES

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

For each area of review, the following review procedures apply:

1. Loading Combinations, System Operating Transient, and Stress Limits

The objectives in reviewing the loading combinations and stress limits employed by the applicant in the design of Code Class 1, 2, and 3 components, component supports, and Class CS core support structures are to confirm that the appropriate postulated events have been included, that the loading combinations (including system operating transients) and the designation of design and service stress limits are appropriate. The review conducted during the CP stage determines that the objectives have been addressed and are being implemented in the design by obtaining a commitment from the applicant that specific design criteria will be utilized.

By checking selected Code required Design Documents such as Design Reports, Load Capacity Data Sheets, and related material, the OL stage review verifies that the design criteria have been utilized and that components have been designed to meet the objectives. To assure that these objectives are met, the review is performed as follows:

a. The applicant's proposed design and service loadings, and combinations thereof, are reviewed for completeness and for appropriate designation of corresponding design and service stress limits.

b. The combination of design and service loadings, including procedures for combination, proposed by the applicant for each Code-constructed item are reviewed to determine if they are adequate. This aspect of the review is made by comparison with the loading combinations and procedures for combination set forth in Appendix A. Deviations from the position are evaluated on a case-by-case basis by questions addressed to the applicant to determine the rationale and justification for exceptions. Final determination is based on engineering judgment and past experience with prior applications.

c. The design and service stress limits selected by the applicant for each set of design and service loading combinations as established in (a) are reviewed to determine if they meet those specified in Appendix A. The provisions for piping component functional capability are reviewed to determine their adequacy in meeting the objectives set forth in Appendix A. Deviations from the position may be permitted provided justification is presented by the applicant. The acceptability determination is based on considerations of adequate margins of safety.

2. Design and Installation of Pressure Relief Devices

The objective of the review of the design and installation of pressure relief devices is to assure the adequacy of the design and installation so that there is assurance of the integrity of the pressure relieving devices and associated piping during the functioning of one or more of the relief devices. In the CP review, it is determined whether there is reasonable assurance that the final design will meet these objectives. At the OL stage, the final design is reviewed to determine that the objectives have been met.

The review is performed as follows:

a. The design of the pressure retaining boundary of the device is reviewed by comparison with the Code. Since explicit rules are not yet available within the Code for the design of safety and pressure relief valves, the design is reviewed on the basis of reference to sections of the Code on vessels, piping, and line valves, and ASME Code Case N-100 (Reference 6).

   Allowable stress limits are compared with those in the Code for the appropriate class of construction. Deviations are identified and the applicant is requested to provide justification. Stress limits and loading combinations are covered under the areas entitled "Loading Combinations, System Operating Transients, and Stress Limits" in this SRP section.

b. The design of the installation is reviewed for structural adequacy to withstand the dynamic effects of relief valve operation. The applicant should include and discuss: reaction force, valve opening sequence, valve opening time, method of analysis, and magnitude of a dynamic load factor (if used). In reaching an acceptance determination, the reviewer compares the submission with the requirements in subsection II.2 of this SRP section.

Where deviations occur, they are identified and the justification is evaluated. Valve opening sequence effects must consider the worst combination possible and forcing functions must be justified with valve opening time data. The review is based in part on comparisons with prior acceptable designs tested in operating plants.

3. Component Supports

The objective in the review of component supports is to determine that adequate attention has been given the various aspects of design and analysis, so that there is assurance as to structural integrity of supports and as to operability of active components that interact with component supports.

The reviewer should be assured that the applicant's PSAR contains discussions and commitments to develop and utilize a snubber operability assurance program containing the elements specified in paragraphs (1) through (6) of subsection II.3.b of this SRP section. A commitment to provide in the FSAR the information specified in paragraph (7) of subsection II.3.b of this SRP section is sufficient for the CP review stage. During the OL review the FSAR should contain summaries in sufficient detail to verify the PSAR commitments.

The structural integrity of the three types of component supports described in subsection I.3 of this SRP section are reviewed against the criteria and guidelines of subsection II.3 of this SRP section.

IV. EVALUATION FINDINGS

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

The staff concludes that the specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 pressure retaining components are acceptable and meet the requirements of 10 CFR Part 50, §50.55a and General Design Criteria 1, 2, 4, 14, and 15. This conclusion is based on the following:

1. The applicant met the requirements of 10 CFR Part 50, §50.55a and General Design Criteria 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components by insuring that systems and components important to safety are designed to quality standards commensurate with their importance to safety and that these systems can accommodate the effects of normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from earthquakes. The specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 pressure retaining components in

systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

2. The applicant has met the requirements of 10 CFR Part 50, §50.55a and General Design Criteria 1, 2, and 4 with respect to the criteria used for design and installation of ASME Code Class 1, 2, and 3 overpressure relief devices by insuring that safety and relief valves and their installations are designed to standards which are commensurate with their safety functions, and that they can accommodate the effects of discharge due to normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from the safe shutdown earthquake. The relevant requirements of General Design Criteria 14 and 15 are also met with respect to assuring that the reactor coolant pressure boundary design limits for normal operation including anticipated operational occurrences are not exceeded. The criteria used by the applicant in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

3. The applicant has met the requirements of 10 CFR Part 50, §50.55a and General Design Criteria 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports by insuring that component supports important to safety are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from the safe shutdown earthquake. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, has met the positions and criteria of Regulatory Guides 1.124 and 1.130 and are in accordance with NUREG-0484 and NUREG-0609. The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity.

Class CS component evaluation findings are covered in SRP Section 3.9.5 in connection with reactor internals.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

1. 10 CFR Part 50, §50.55a, "Codes and Standards."

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (a) General Design Criterion 1, "Quality Standards and Records," (b) General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," (c) General Design Criterion 4, "Environmental and Missile Design Bases," (d) General Design Criterion 14, "Reactor Coolant Pressure Boundary," and (e) General Design Criterion 15, "Reactor Coolant System Design."

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

4. Standard Review Plan Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment Important to Safety."

5. Appendix A to SRP Section 3.9.3, "Stress Limits for ASME Class 1, 2, and 3 Components and Component Supports of Safety-Related Systems and Class CS Core Support Structures Under Specified Service Loading Combinations."

6. ASME Code Case N-100, "Pressure Relief Valve Design Rules, Section III, Division 1, Class 1, 2 and 3."

7. Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports."

8. Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports."

9. NUREG-0484, "Methodology for Combining Dynamic Loads."

10. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems."

APPENDIX A

STANDARD REVIEW PLAN SECTION 3.9.3
STRESS LIMITS FOR ASME CLASS 1, 2, AND 3 COMPONENTS AND COMPONENT SUPPORTS OF SAFETY-RELATED SYSTEMS AND CLASS CS CORE SUPPORT STRUCTURES UNDER SPECIFIED SERVICE LOADING COMBINATIONS

A. INTRODUCTION

Nuclear power plant components and supports are subjected to combinations of loadings derived from plant and system operating conditions, natural phenomena, postulated plant events, and site-related hazards. Section III, Division 1 of the ASME Code (hereafter referred to as the Code) provides specific sets of design and service stress limits that apply to the pressure retaining or structural integrity of components and supports when subjected to these loadings. The design and service stress limits specified by the Code do not assure, in themselves, the operability of components, including their supports, to perform the mechanical motion required to fulfill the component's safety function. Certain of the service stress limits specified by the Code (i.e., level C and D) may not assure the functional capability of components, including their supports, to deliver rated flow and retain dimensional stability. Since the combination of loadings, the selection of the applicable design and service stress limits appropriate to each load combination and the proper consideration of operability is beyond the scope of the Code; and the treatment of functional capability, including collapse and deflection limits, is not adequately treated by the Code for all situations, such factors must be evaluated by designers and appropriate information developed for inclusion in the Design Specification or other referenced documents.

Applicants require guidance with regard to the selection of acceptable design and service stress limits associated with various loadings and combinations thereof, resulting from plant and system operating conditions and design basis events, natural phenomena, and site-related hazards. The relationship and application of the terms "design conditions," "plant operating conditions," "system operating conditions," and the formerly used term "component operating conditions," now characterized by four levels of service stress limits, have not been clearly understood by applicants and their subcontractors.

For example, under the "faulted plant or system condition" (e.g., due to LOCA within the reactor coolant pressure boundary), the emergency core cooling system (ECCS) should be designed to operate and deliver rated flow for an extended period of time to assure the safe shutdown of the plant. Although the "plant condition" is termed "faulted," components in the functional ECCS must perform the safety function under a specified set of service loadings which includes those resulting from the specified plant postulated events. The selection of level "D" (related to the "faulted" condition) service stress limits for this system, based solely on the supposition that all components may use this limit for a postulated event resulting in the faulted plant condition cannot be justified, unless system operability is also demonstrated.

This appendix is necessary to improve consistency and understanding of the basic approach in the selection of load combinations applicable to safety-related systems and to establish acceptable relationships between plant postulated events, plant and system operating conditions, component and component support design, and service stress limits, functional capability, and operability.

B. DISCUSSION

Current reviews of both standardized plants and custom plants have indicated the need for additional guidance to reach acceptable design conclusions in the following areas:

(1) Relationship between certain plant postulated events, plant and system operating conditions, resulting loads and combinations thereof, and appropriate design and service stress limits for ASME Class 1, 2 and 3 components and component supports and Class CS core support structures.

(2) Relationship of component operability assurance, functional capability, and allowable design and service stress limits for ASME Class 1, 2, and 3 components and component supports.

The Code provides five categories of limits applicable to design and service loadings (design, level A, level B, level C, and level D). The Code rules provide for structural integrity of the pressure retaining boundary of a component and its supports, but specifically exclude the subject of component operability and do not directly address functional capability. The types of loadings to be taken into account in designing a component are specified in the Code, but rules specifying how the loadings, which result from postulated events and plant and system operating conditions, are to be combined and what stress level is appropriate for use with loading combinations are not specified in the Code. It is the responsibility of the designer to include all this information in the Code required Design Specification of each component and support.

C. POSITION

Effective with the 1977 Edition, the Code provides design stress limits and four sets of service stress limits for all classes of components, component supports, and core support structures. The availability of such design and service stress limits within the Code requires that the MEB review and determine maximum acceptable design and service stress limits which may be used with specified loads, or combinations thereof, for components and component supports of safety-related systems (refer to definition in Table III) and core support structures.

This appendix provides guidance for dealing with the components and component supports of safety-related systems and core support structures in the following areas:

(1) Consideration of design loadings and limits.

(2) Consideration of service loading combinations resulting from postulated events and the designation of acceptable service limits.

(3) Consideration of piping functional capability and operability of active pumps and valves under service loading combinations resulting from postulated events.

(4) Applicability of the appendix to components, component support structures, and core support structures and procedures for compliance.

1.0 ASME CLASS 1, 2, AND 3 COMPONENTS AND COMPONENT SUPPORTS OF SAFETY-RELATED SYSTEMS AND CLASS CS CORE SUPPORT STRUCTURES

1.1 Design Considerations and Design Loadings

ASME Code Class 1, 2, and 3 components, component supports, and class CS core support structures shall be designed to satisfy the appropriate subsections of the Code in all respects, including limitations on pressure, and the requirements of this appendix. Component supports that are intended to restrain either force and displacement or anchor movement shall be designed to maintain deformations within appropriate limits as specified in the component support Design Specifications.

Design loadings shall be established in the Design Specification. The design limits of the appropriate subsection of the Code shall not be exceeded for the design loadings specified.

1.2 Service Loading Combinations

The identification of individual loads and the appropriate combination of these loads (i.e., sustained loads, loads due to system operating transients SOT, OBE, SSE, LOCA, DBPB, MS/FWPB and their dynamic effects) shall be in accordance with Section 1.3. The appropriate method of combination of these loads shall be in accordance with NUREG-0484, "Methodology for Combining Dynamic Loads" (Reference 9).

1.3 Service Conditions

1.3.1 Service Limit A

Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level A when subjected to sustained loads resulting from normal plant/system operation.

1.3.2 Service Limit B

Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level B when subjected to the appropriate combination of loadings resulting from (1) sustained loads, (2) specified plant/system operating transients (SOT), and (3) the OBE.

1.3.3 Service Limit C

(a) Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level C when subjected to the appropriate combination of loadings resulting from (1) sustained loads, and (2) the DBPB.

(b) The DBPB includes loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

1.3.4 Service Limit D

(a) Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service stress limit not greater than Level D when subjected to the appropriate combination of loadings resulting from (1) sustained loads, (2) either the DBPB, MS/FWPB, or LOCA, and (3) and SSE.

(b) The DBPB, MS/FWPB, and LOCA include loads from the postulated pipe breaks, themselves, and also any associated system transients or dynamic effects resulting from the postulated pipe breaks. Asymmetric blowdown loads on PWR primary systems shall be incorporated per NUREG-0609 (Reference 10).

2.0 OPERABILITY AND FUNCTIONAL CAPABILITY

2.1 Active Pumps and Valves

SRP Section 3.10 (Reference 4) shall demonstrate that the pump or valve, as supported, can adequately sustain the designated combined service loadings at a stress level at least equal to the specified service limit, and can perform its safety function without impairment. Loads produced by the restraint of free end displacement and anchor point motions shall be included.

2.2 Snubbers

The operability requirements specified for mechanical and hydraulic snubbers installed on safety-related systems is subject to review by the staff. When snubbers are used, their need shall be clearly established and their design criteria presented.

2.3 Functional Capability

The design of Class 1, 2, and 3 piping components shall include a functional capability assurance program. This program shall demonstrate that the piping components, as supported, can retain sufficient dimensional stability at service conditions so as not to impair the system's functional capability. The program may be based on tests, analysis, or a combination of tests and analysis.

3.0 TABLES

3.1 Table I summarizes the requirements of this appendix for use with ASME Class 1, 2, and 3 components, component supports, and Class CS core support structures. The table illustrates plant events, system operating conditions, service loading combinations, and service stress limits and should always be used in conjunction with the text of this appendix.

3.2 Table II defines all the terms used in this appendix.

4.0 PROCEDURES FOR COMPLIANCE

4.1 Design Specification and Safety Analysis Report

(a) The dasign options provided by the Code and related design criteria specified in the Code required Design Specification for ASME Class 1, 2, and 3 components, component supports, and Class CS core support structures should be summarized in sufficient detail in the Safety Analysis Report of the application to permit comparison with this Appendix.

(b) The presentation in the PSAR should specify and account for all design and service loadings, method of combination, the designation of the appropriate design and service stress limits (including primary and secondary stresses, fatigue consideration, and special limits on pressure when appropriate) for each loading combination presented, and the provisions for functional capability.

(c) The presentation in the FSAR should indicate how the criteria in Sections 1 and 2 of this appendix have been implemented.

(d) The staff may request the submission of the Code required Design Documents such as Design Specifications, Design Reports, Load Capacity Data Sheets, or other related material or portions thereof to establish that the design criteria, the analytical methods, and functional capability satisfy the guidance provided by this appendix. This may include information provided to, and received from, component and support manufacturers. As an alternative to the applicant submitting these documents, the staff may require them to be made available for review at the applicant's or vendor's office.

4.2 Use with Regulatory Guides

The information and requirements contained in this appendix supersede those in the October 1973 version of Regulatory Guide 1.67 and the May 1973 version of Regulatory Guide 1.48. Regulatory Guides 1.124 and 1.130 on Class 1 linear and Class 1 plate and shell component support structures are to be supplemented by this appendix.

TABLE I

Allowable Service Stress Limits for Specified Service Loading Combinations for ASME Class 1 Components and Class CS Support Structures

	Plant Event[2]	System Operating Conditions	Service Loading Combination[1]	Service Stress Limit
1.	Normal Operation	Normal	Sustained Loads	A
2.	Plant/System Operating Transients (SOT) + OBE	Upset	Sustained Loads + SOT + OBE	B[3]
3.	DBPB	Emergency	Sustained Loads + DBPB	C[3]
4.	MS/FWPB	Faulted	Sustained Loads + MS/FWPB	D[3]
5.	DBPB or MS/FWPB + SSE	Faulted	Sustained Loads + DBPB or MS/FWPB + SSE	D[3]
6.	LOCA	Faulted	Sustained Loads + LOCA	D[3]
7.	LOCA + SSE	Faulted	Sustained Loads + LOCA + SSE	D[3]

NOTE: [1]The appropriate method of combination is subject to review and evaluation. Refer to Section 1.2.

[2]Refer to Table II for definition of terms.

[3]In addition to meeting the specified service stress limits for given load combinations, operability and functional capability must also be demonstrated as discussed in subsection 2.0 of this appendix and in SRP Section 3.10.

TABLE II

DEFINITION OF TERMS

Active Pumps and Valves - A pump or valve which must perform a mechanical motion in order to shut down the plant or mitigate the consequences of a postulated event. Safety and relief valves are specifically included.

Component and Support Functional Capability - Ability of a component, including its supports, to deliver rated flow and retain dimensional stability when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

Component and Support Operability - Ability of an active component, including its support, to perform the mechanical motion required to fulfill its designated safety function when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

DBPB - Design Basis Pipe Breaks - Those postulated pipe breaks other than a LOCA or MS/FWPB. This includes postulated pipe breaks in Class 1 branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.

This condition includes loads from the postulated pipe breaks, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

Design Limits - The limits for the design loadings provided in the appropriate subsection of Section III, Division 1, of the ASME Code.

Design Loads - Those pressures, temperatures, and mechanical loads selected as the basis for the design of a component.

Functional System - That configuration of components which, irrespective of ASME Code Class designation or combination of ASME Code Class designations, performs a particular function (i.e., each emergency core cooling system performs a single particular function and yet each may be comprised of some components which are ASME Class 1 and other components which are ASME Code Class 2).

LOCA - Loss-of-Coolant Accidents - Defined in Appendix A of 10 CFR Part 50 as "those postulated accidents that result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system."

This condition includes the loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

MS/FWPB - Main Steam and Feedwater Pipe Breaks - Postulated breaks in the main steam and feedwater lines. For a BWR plant this may be considered as a LOCA event depending on the break location.

This condition includes the loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

OBE - Operating Basis Earthquake - Defined in Section III (d) of Appendix A of 10 CFR Part 100 as "that earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant, necessary for continued operation without undue risk to the health and safety of the public, are designed to remain functional."

This condition includes the loads from the postulated seismic event, itself, and also any associated system transients or dynamic effects resulting from the postulated seismic event.

Piping Components - These items of a piping system such as tees, elbows, bends, pipe and tubing, and branch connections constructed in accordance with the rules of Section III of the ASME Code.

Postulated Events - Those postulated natural phenomena (i.e., OBE, SSE), postulated site hazards (i.e., nearby explosion), or postulated plant events (i.e., DBPB, LOCA, MS/FWPB) for which the plant is designed to survive without undue risk to the health and safety of the public. Such postulated events may also be referred to as design basis events.

SSE - Safe Shutdown Earthquake - Defined in Section III(c) of Appendix A of 10 CFR Part 100 as "that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is the earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure:

(1) The integrity of the reactor coolant pressure boundary.

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline."

This condition includes the loads from the postulated seismic event, itself, and also any associated system transients or dynamic effects resulting from the postulated seismic event.

Service Limits - The four limits for the service loading as provided in the appropriate subsection of Section III, Division 1, of the ASME Code.

Service Loads - Those pressure, temperature, and mechanical loads provided in the Design Specification.

SOT - System Operating Transients - The transients and their resulting mechanical responses due to dynamic occurrences caused by plant or system operation.

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3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

The MEB reviews the following areas of the applicant's safety analysis report (SAR) that cover the inservice testing of certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter "the Code"). Other pumps and valves not categorized as Code Class 1, 2, or 3 may be included if they are considered to be safety related by the staff. Compliance with the Code will assure conformance with 10 CFR Part 50, Appendix A, General Design Criteria 37, 40, 43, 46, 54, and 10 CFR Part 50, §50.55a(g);

1. Inservice Testing of Pumps

   a. The descriptive information in the SAR covering the inservice test program is reviewed for those ASME Code Class 1, 2, and 3 system pumps whose function is required for safety, and in addition includes pumps not categorized as Code Class 1, 2, or 3 but which are considered to be safety related.

   b. Procedures for testing for speed, fluid pressure, flow rate, vibration amplitude, lubricant level or pressure, and bearing temperature at normal pump operating conditions are reviewed.

   c. The pump test schedule is reviewed.

   d. The methods described in the SAR for measuring the reference values and inservice values for the pump parameters above are reviewed.

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2. Inservice Testing of Valves

The descriptive information in the SAR covering the inservice test program is reviewed for those ASME Code Class 1, 2, and 3 valves whose function is required for safety. This review does not include those nonsafety-related valves exempted by the Code.

3. Relief Requests

10 CFR Part 50, §50.55a(g) requires a nuclear power facility to periodically update its inservice testing program to meet the requirements of future revisions of Section XI of the ASME Code. However, if it proves impractical to implement these criteria, the applicant is allowed to submit requests for relief from Section XI requirements on a case-by-case basis. Accordingly, any requests for relief are reviewed by the staff to determine if the proposed exceptions to Section XI will degrade the overall plant safety. Due consideration is given to the burden upon the applicant that could result if the criteria of Section XI were imposed on the facility.

II. ACCEPTANCE CRITERIA

The acceptance criteria is based on meeting the relevant requirements set forth in General Design Criteria 37, 40, 43, 46, 54, and 10 CFR Part 50, §50.55a(g). The relevant requirements are as follows:

A. General Design Criterion 37, as it relates to periodic functional testing of the emergency core cooling system to assure the leak tight integrity and performance of its active components.

B. General Design Criterion 40, as it relates to periodic functional testing of the containment heat removal system to assure the leak tight integrity and performance of its active components.

C. General Design Criterion 43, as it relates to periodic functional testing of the containment atmospheric cleanup systems to assure the leak tight integrity and the performance of the active components, such as pumps and valves.

D. General Design Criterion 46, as it relates to periodic functional testing of the cooling water system to assure the leak tight integrity and performance of the active components.

E. General Design Criterion 54 as it relates to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation and determine valve leakage acceptability.

F. 10 CFR Part 50, §50.55a(g), as it relates to including pumps and valves whose function is required for safety in the Inservice Inspection Program to verify operational readiness by periodic testing.

Specific criteria necessary to meet the relevant requirements of the Commission regulations identified above are as follows:

1. Inservice Testing of Pumps

   a. The scope of the applicant's test program is acceptable if it is in agreement with IWP-1000 of Section XI of the Code and in addition includes pumps not categorized as Code Class 1, 2, or 3, but which are considered to be safety related. Since the pump test program is based on the detection of changes in the hydraulic and mechanical condition of a pump relative to a reference test specified in IWP-3000, the establishment of a reference set of parameters and a consistent test method is a basic criterion of the program.

   b. The pump test program is acceptable if it meets the requirements for establishing reference values and the periodic testing schedule of IWP-3000 of Section XI of the Code. The allowable ranges of inservice test quantities, corrective actions, and bearing temperature tests are established by IWP-3000 and IWP-4000. The pump test schedule in the plant technical specification is required to comply with these rules.

   c. The test frequencies and durations are acceptable if the provisions of IWP-3000 of Section XI of the Code are met.

   d. The methods of measurement are acceptable if the test program meets the requirements of IWP-4000 of Section XI of the Code with regard to instruments, pressure measurements, temperature measurements, rotational speed, vibration measurement, and flow measurements.

2. Inservice Testing of Valves

   a. To be acceptable, the SAR valve test list must contain all safety-related Code Class 1, 2, and 3 valves required by IWV-1100 except those nonsafety-related valves exempted by the Code and in addition includes valves not categorized as Code Class 1, 2, or 3 but which are considered safety related. The SAR valve list must include a valve categorization which complies with the provisions of IWV-2000 of Section XI of the Code. Each specific valve to be tested by the rules of Subsection IWV is listed in the SAR by type, valve identification number, code class, and IWV-2000 valve category.

   b. The valve test procedures are acceptable if the provisions of IWV-3000 of Section XI of the Code are met with regard to preservice and periodic inservice valve testing.

3. Information Required for Review of Relief Requests

   a. Identify component for which relief is requested:

      (1) Name and number as given in FSAR
      (2) Function
      (3) ASME Section III Code Class
      (4) For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000.

   b. Specifically identify the ASME Code requirement that has been determined to be impractical for each component.

c. Provide information to support the determination that the requirement in item (b) is impractical; i.e., state and explain the basis for requesting relief.

d. Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.

e. Provide an explanation as to why the proposed inservice testing will provide an acceptable level of quality and safety and not endanger the public health and safety.

f. Provide the schedule for implementation of the procedure(s) in item (d).

Requests for relief from Section XI requirements will be granted by the staff if the applicant has adequately demonstrated either of the following:

a. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety, and noncompliance will provide an acceptable level of quality and safety.

b. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case. For each area of review, the following review procedures are followed:

1. Inservice Testing of Pumps

   a. The scope of the applicant's program is reviewed for agreement with subsection II.1.a. The program is acceptable if a preservice test program is used to establish reference values. The periodic inservice program must verify the reference values within acceptable limits.

   b. The pump test program procedures must agree with the requirements of subsection II.1.b. The program is best presented in tabular form.

   c. The inservice test frequencies and test durations are reviewed for agreement with subsection II.1.c.

   d. The test procedures described in the SAR are reviewed for agreement with subsection II.1.d. The SAR need only provide the necessary information to permit a conclusion that the methods of measurement and the data acquisition system will provide the needed data. The reviewer does not approve or disapprove the instruments or methods proposed or used.

2. Inservice Testing of Valves

   a. The SAR valve test list and categorization are reviewed for agreement with Subsection II.2.a.

b. The valve test program is acceptable if the procedures follow the rules of subsection II.2.b for preservice and periodic inservice testing.

3. Relief Requests

Requests for relief from Section XI requirements are reviewed to determine that sufficient information has been provided and that the acceptance criteria of subsection II.3 have been met.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided in accordance with the requirements of this SRP section and that his evaluation supports a conclusion of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the applicant's pumps and valves test program is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 37, 40, 43, 46, 54, and §50.55a(g). This conclusion is based on the applicant having provided a test program to ensure that safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant. This program includes baseline preservice testing and periodic inservice testing. The program provides for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress. Applicant has also formulated his inservice test program to include all safety-related Code Class 1, 2, and 3 pumps and valves and to include those pumps and valves which are not Code Class 1, 2, and 3 but are considered to be safety related.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

2. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

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

4. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water Systems."

5. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

6. ASME Boiler and Pressure Vessel Code, Section III and Section XI, Subsections IWP and IWV, American Society of Mechanical Engineers.

7. Code of Federal Regulations, Title 10, Part 50, Section 50.55a, "Codes and Standards."

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SECTION 6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 36, "Inspection of Emergency Core Cooling System"; Criterion 39, "Inspection of Containment Heat Removal System"; Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems"; and Criterion 45, "Inspection of Cooling Water System," of Appendix A to 10 CFR Part 50 require that the subject systems be designed to permit appropriate periodic inspection of important component parts to assure system integrity and capability. General Design Criterion 37, "Testing of Emergency Core Cooling System"; Criterion 40, "Testing of Containment Heat Removal System"; Criterion 43, "Testing of Containment Atmosphere Cleanup Systems"; and Criterion 46, "Testing of Cooling Water System," require in part that the subject systems be designed to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of their components.

Inservice inspection programs are based on the general requirements of 10 CFR Part 50, Section 50.55a, as detailed in Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components." Inservice inspection includes a preservice inspection prior to initial plant startup.

The following areas relating to the inservice inspection (ISI) program for NRC Quality Group B and C (ASME Boiler and Pressure Vessel Code, Section III, Code Class 2 and 3) components are reviewed:

1. Components Subject to Examination

   The descriptive information in the applicant's or licensee's safety analysis report (SAR) is reviewed to establish that all the ASME Boiler and Pressure Vessel Code (hereinafter "the Code"), Section III, Article NA-2000, Class 2 and Class 3 components are included in the ISI program. The Mechanical

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Engineering Branch verifies in SRP Section 3.2.2 that the systems classified as Code Class 2 and 3 agree with Article NA-2000 of Section III and with the definitions of the General Design Criteria. The inservice inspection requirements for ASME Code Class 1 components in the reactor coolant pressure boundary and steam generator tubes are reviewed by MTEB as part of its primary review responsibility for SRP Sections 5.2.4 and 5.4.2.2, respectively.

2. Accessibility

The descriptive information, including drawings, is reviewed by the MTEB to establish that the Code Section XI, Subarticle IWA-1500, provisions for system accessibility are included in the applicant's or licensee's layout and design of these systems.

3. Examination Categories and Methods

The required examination categories and methods included in IWC-2000 and IWD-2000 of Section XI are reviewed.

4. Inspection Intervals

The required examinations and inspections listed in the SAR are reviewed and compared to the requirements in IWC-2000 and IWD-2000 of Section XI to verify that they will be performed within the designated inspection interval.

5. Evaluation of Examination Results

The information concerning repair procedures is reviewed for compliance with Articles IWC-4000 and IWD-4000 of Section XI. The information concerning evaluation of examination results is reviewed for compliance with IWC-3000 and IWD-3000 of the Code. If these requirements are in course of preparation in the applicable Code edition for a program, suitable alternative provisions, such as the requirements in IWB-3000 or those in later approved editions of the Code, should be proposed by the applicant or licensee.

6. System Pressure Tests

The pressure test program is reviewed for compliance with Articles IWC-5000 and IWD-5000 of Section XI to establish that leakage and signs of structural distress are inspected for on a periodic basis.

7. Augmented ISI to Protect Against Postulated Piping Failures

The augmented inservice inspection program as specified in SRP Section 3.6.1 to provide assurance against postulated piping failures of high-energy fluid systems between containment isolation valves is reviewed.

8. Code Exemptions

The ASME Section XI Code exemptions as permitted by IWC-1220 are reviewed.

9. Relief Requests

Requests for relief from the ASME Code Section XI examination requirements which are found to be impractical due to the limitations of design, geometry, or materials of construction of components are evaluated in accordance with Section 50.55a of 10 CFR Part 50.

II. ACCEPTANCE CRITERIA

The requirements for periodic inspection and testing of Class 2 and 3 systems in General Design Criteria 36, 37, 39, 40, 42, 43, 45, and 46 are specified in part in 10 CFR Part 50, Section 50.55a, "Codes and Standards," and detailed in Section XI of the ASME Code. Compliance with the preservice and inservice examinations of 10 CFR Part 50, Section 50.55a, as detailed in Section XI of the Code, constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 36, 37, 39, 40, 42, 43, 45, and 46. Specific acceptance criteria for meeting the ISI requirements of these General Design Criteria and 10 CFR Part 50, Section 50.55a for the areas of review described in subsection I of this SRP section are as follows:

1. Components Subject to Inspection

The applicant's or licensee's definition of Code Class 2 and 3 components and systems subject to an ISI program is acceptable if it is in agreement with the definitions of Code Section III, Article NA-2000. The interpretation of code classifications by the applicant or licensee is subject to review by the Mechanical Engineering Branch in SRP Section 3.2.2 for compliance with safety criteria pertaining to component classification. (Refer to NA-2000 of Section III.)

2. Accessibility

The design and arrangement of Class 2 and 3 systems are acceptable if the applicant or licensee includes allowances for adequate clearances to conduct the examinations specified in IWC-2000 and IWD-2000 at the frequency specified. Special design considerations are given to those systems that are intended to be examined during normal reactor operation.

3. Examination Categories and Methods

The examination categories and requirements specified in the SAR are acceptable if in agreement with the criteria of IWC-2000 and IWD-2000 of the Code. Every area subject to examination should fall within one or more of the examination categories and must be examined at least to the extent specified. The methods of examination for the components are also listed in the requirements of IWC-2000 and IWD-2000 of the Code.

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

a. The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with IWA-2000 of the Code.

b. Alternative examination methods, combination of methods, or newly developed techniques to those given in a. above are acceptable provided that the results are equivalent or superior. The acceptance standards for these alternate methods are given in Section XI, IWC-3000 and IWD-3000.

4. Inspection Intervals

The inservice inspection program schedule given in the SAR is acceptable if the required examinations are completed during each ten-year interval, hereinafter designated as the inspection interval, and as required by Articles IWC-2000 and IWD-2000 of Section XI.

5. Evaluation of Examination Results

The methods for evaluation of examination results are reviewed for compliance with Articles IWC-3000 and IWD-3000 in the Code. If the applicable edition of the Code states that these articles are in the course of preparation, the rules of IWB-3000 shall apply.

6. System Pressure Tests

The SAR program for Class 2 and 3 system pressure testing is acceptable if it meets the criteria of IWC-5000 and IWD-5000 of Section XI.

7. Augmented ISI to Protect Against Postulated Piping Failures

High-energy fluid system piping between containment isolation valves should receive an augmented ISI as follows:

a. Protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the Code, Section XI, Division 1.

b. For those portions of high energy fluid system piping between containment isolation valves, the extent of inservice examination completed during each inspection interval should provide 100% volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

c. For those portions of high-energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

d. The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Article IWC-2000.

8. Code Exemptions

The applicant or licensee should list the exemptions from Code examination requirements that have been permitted by IWC-1220 of the Code.

9. Relief Requests

Request for relief from the ASME Code Section XI examination requirements which are found to be impractical due to the limitations of design, geometry, or materials of construction of components are evaluated in accordance with Section 50.55a of 10 CFR Part 50.

III. REVIEW PROCEDURES

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

For each area of review the following review procedure is followed:

1. Components Subject to Inspection

The applicant's or licensee's components and system classifications under Class 2 are reviewed for agreement with subsection II.1 of this SRP section. The interpretation of Code classifications is the responsibility of the Mechanical Engineering Branch in the review of SRP Section 3.2.2, should a discrepancy occur between the SAR and subsection II.1 of this SRP section.

The applicant's or licensee's classification of Class 3 systems is reviewed for agreement with subsection II.1 of this SRP section. Any safety-related, fluid-carrying components not included in Class 1 or Class 2 and not a part of the containment structure are included in Class 3.

2. Accessibility

The design and arrangement of Class 2 and 3 systems are reviewed in terms of accessibility for ISI to establish that the design meets the requirements of subsection II.2 of this SRP section. No remote inspection program is required for Code Class 2 or 3 components.

3. Examination Categories and Methods

The reviewer verifies that the examination categories and methods as described by the SAR are the same as those specified in subsection II.3 of this SRP section.

4. Inspection Intervals

The inservice inspection program for Class 2 and 3 components in the plant technical specifications is reviewed to establish that each area and component in the program is inspected on a schedule in agreement with subsection II.4 of this SRP section.

5. Evaluation of Examination Results

The reviewer verifies that the evaluation of examination results described in the SAR is in accordance with subsection II.5 of this SRP section.

6. <u>System Pressure Test</u>

The system pressure test program is acceptable if it meets the criteria of subsection II.6 of this SRP section.

7. <u>Augmented ISI to Protect Against Postulated Piping Failures</u>

The reviewer verifies that the augmented inservice inspection program as described in the SAR meets the acceptance criteria identified in subsection II.7 of this SRP section.

8. <u>Code Exemptions</u>

The reviewer verifies that the exemptions from Code examinations are in accordance with the criteria in IWC-1220.

9. <u>Relief Requests</u>

The reviewer determines if the applicant or licensee has demonstrated that a code requirement is impractical due to the limitations of design, geometry, or materials of construction of components.

IV. EVALUATION FINDINGS

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

To ensure that no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones are inspected prior to reactor startup and periodically throughout the life of the plant. In addition, Code Class 2 and 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

The applicant (licensee) has stated that his inservice inspection (ISI) program will comply (complies) with the rules published in 10 CFR Part 50, Section 50.55a, and Section XI of the ASME Code, (   ) Edition, including addenda through the (   ) Addenda. The ISI program will consist of a preservice inspection plan and an inservice inspection plan.

Examples of Code Class 2 systems are: residual heat removal systems, portions of chemical and volume control systems (in PWR plants), portions of control rod drive systems (in BWR Plants), and engineered safety features not part of Code Class 1 systems. Examples of Code Class 3 systems are: component cooling water systems and portions of radwaste systems. All of these systems transport fluids.

The staff concludes that the inservice inspection program is acceptable and meets the inspection and pressure testing requirements of General Design Criteria 36, 37, 39, 40, 42, 43, 45, and 46 and 10 CFR Part 50, Section 50.55a. This conclusion is based on the applicant's or licensee's meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, as reviewed by the staff and determined to be appropriate for this application.

V. IMPLEMENTATION

The following is intend provide guidance to applicants and licensees regarding the NRC staf s pian for using this SRP section. Except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the Commission's regulations, the methods described herein will be used by the staff in its evaluation of conformance with Commission regulations. Implementation schedules are defined in Section 50.55a of 10 CFR Part 50.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System"; Criterion 37, "Testing of Emergency Core Cooling System"; Criterion 39, "Inspection of Containment Heat Removal System"; Criterion 40, "Testing of Containment Heat Removal System"; Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems"; Criterion 43, "Testing of Containment Atmosphere Cleanup Systems"; Criterion 45, "Inspection of Cooling Water Systems"; and Criterion 46, "Testing of Cooling Water System."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NA-2000 and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

10.3 MAIN STEAM SUPPLY SYSTEM

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

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

The MSSS for the BWR direct cycle plant extends from the outermost containment isolation valves up to and including the turbine stop valves, and includes connected piping of 2-1/2 inches nominal diameter and larger up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of reactor operation. The MSSS for the PWR indirect cycle plant extends from the connections to the secondary sides of the steam generators up to and including the turbine stop valves, and includes the containment isolation valves, safety and relief valves, connected piping of 2-1/2 inches nominal diameter and larger up to and including the first valve that is either normally closed or capable of automatic closure during all modes of operation and the steam line to the auxiliary feedwater pump turbine. The ASB is responsible for the review of the MSSS from the containment up to and including the outermost isolation valve. The PSB is responsible for the review of the remainder of the MSSS. (The turbine stop valve review is included in SRP Section 10.2.) The PSB also determines the adequacy of the design, installation, inspection, and testing of the electrical power supplies for essential components required for proper operation of the MSSS. The design of the MSSS must be in accordance with General Design Criteria 2, 4, 5, and 34.

Enclosure 8

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

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

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

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

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

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

   f. The system design capability can withstand adverse dynamic loads, such as steam hammer resulting from rapid valve closure and relief valve fluid discharge loads.

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

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

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

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

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

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

4. ASB also performs the following reviews under the SRP sections indicated:

a. Review for flood protection is performed under SRP Section 3.4.1.

b. Review of the protection against internally generated missiles is performed under SRP Section 3.5.1.1.

c. Review of the structures, systems, and components to be protected against externally generated missiles is performed under SRP Section 3.5.2.

d. Review of high- and moderate-energy pipe breaks is performed under SRP Section 3.6.1.

In the review of the main steam supply system, the ASB and PSB will coordinate other branches' evaluations that interface with the overall review of the system as follows: The Reactor Systems Branch (RSB) identifies essential components associated with the portion of the MSSS inside the primary containment that are required for normal operations and accident conditions, establishes shutdown cooling load requirements versus time, and verifies the design transient used in establishing the flow capacity and setpoint(s) of steam generator relief and safety valves as part of its primary review responsibility for SRP Section 5.2. The Structural and Geotechnical Engineering Branch (SGEB) determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles as part of its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5. The Equipment Qualification Branch (EQB) reviews the seismic and environmental qualification of components under SRP Sections 3.10 and 3.11. The Mechanical Engineering Branch (MEB) determines that the components, piping, and supports are designed in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 through 3.9.3. The MEB determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2. The MEB also reviews the adequacy of the inservice testing program of the system valves as part of its primary review responsibility for SRP Section 3.9.6. The Materials Engineering Branch (MTEB) verifies, upon request, the compatibility of the materials of construction with service conditions. The Instrumentation and Control Systms Branch (ICSB) reviews portions of the MSSS with respect to the adequacy of design, installation, inspection, and testing of essential components necessary for instrumentation and control functions as part of its primary review responsibility for SRP Sections 7.1, 7.4, 7.5, and 7.7. The Procedures and Systems Review Branch (PSRB) determines the acceptability of the preoperational and startup tests as part of its primary review responsibility for SRP Section 14.0. The reviews for fire protection, technical specifications, and quality assurance are coordinated and performed by the Chemical Engineering Branch, Standardization and Special Projects Branch (SSPB), and Quality Assurance Branch as part of their primary review responsibility for SRP Sections 9.5.1, 16.0, and 17.0, respectively.

For those areas of review identified above as being part of the primary review responsiblity of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections of the corresponding primary branches.

II. ACCEPTANCE CRITERIA

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

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

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

   a. Regulatory Guide 1.29, as related to the seismic design classification of system components, Positions C.1.a, C.1.e, C.1.f, C.2, and C.3.

   b. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles, Appendix Positions 2 and 4.

2. General Design Criterion 4, with respect to safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks, and the position of Regulatory Guide 1.115 as related to the protection of structures, systems, and components important to safety from the effects of turbine missiles, Position C.1.

   The system design should adequately consider steam hammer and relief valve discharge loads to assure that system safety functions can be achieved and should assure that operating and maintenance procedures include adequate precautions to avoid steam hammer and relief valve discharge loads. The system design should also include protection against water entrainment.

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

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

   a. The positions in Branch Technical Position RSB 5-1 as related to the design requirements for residual heat removal.

   b. Issue Number 1 of NUREG-0138 as related to credit being taken for all valves downstream of the main steam isolation valves (MSIV) to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II of this SRP section. For review of operating license (OL) applications, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report.

The procedures for OL applications include a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements for system testing, minimum performance, and surveillance, developed as a result of the SSPB review, as indicated in subsection I of this SRP section.

The primary reviewers, will coordinate this review with the other branches' areas of review as stated in subsection I of this SRP section. The primary reviewers obtain and use such input as required to assure that this review procedure is complete.

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

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

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

3. The SGEB reviews the seismic design bases and MEB reviews the quality and seismic classification as indicated in subsection I of this SRP section. The SAR is reviewed by ASB and PSB to verify that essential portions of the MSSS are designed to Quality Group B and/or seismic Category I requirements, and to verify that the design classifications specified meet the acceptance criteria specified in subsection II of this SRP section. In general:

   a. The main steam lines from the steam generators to the containment isolation valves in PWR plants are classified seismic Category I and Quality Group B.

   b. The main steam lines in BWR plants extending from the outermost containment isolation valve and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operations but not including the turbine stop and bypass valves are classified seismic Category I and a quality group classification in accordance with BTP RSB 3-1.

Alternatively, for BWRs containing a shutoff valve (in addition to the two containment isolation valves) in the MSSS, seismic Category I and a quality group classification in accordance with BTP RSB 3-2 should be applied to that portion of the MSSS extending from the outermost containment isolation valves up to and including the shutoff valve.

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

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

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

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

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

   d. Assure that in the event of a postulated break in a main steam line in a PWR plant, the design will preclude the blowdown of more than one steam generator, assuming a concurrent single active component failure. In this regard, all main steam shut-off valves downstream of the MSIVs, the turbine stop valves, and the control valves are considered to be functional. The reviewer should verify that the main steam isolation valves, shut-off valves in connecting piping, turbine stop valves, and bypass valves can close against maximum steam flow. The reviewer verifies that the SAR provides a tabulation and descriptive text of all flow paths that branch off the main steam lines between the MSIVs and the turbine stop valves. The descriptive information shall include the following for each flow path:

(1) System identification

(2) Maximum steam flow in pounds per hour

(3) Type of shut-off valve(s)

(4) Size of valve(s)

(5) Quality of the valve(s)

(6) Design code of the valve(s)

(7) Closure time of the valve(s)

(8) Actuation mechanism of the valve(s) (i.e., solenoid operated, motor operated, air operated diaphragm valve, etc.)

(9) Motive or power source for the valve actuating mechanism.

e. In the event of a main steam line break, termination of steam flow from all systems identified in d, above, except those that can be used for mitigation of the accident, is required to bring the reactor to a safe cold shutdown. For these systems the reviewer verifies that the SAR describes what design features have been incorporated to assure closure of the steam shut-off valve(s) and what operator actions, if any, are required. If the systems that can be used for mitigation of the accident are not available, or the decision is made to use other means to shut down the reactor, the reviewer verifies that the SAR decribes how these systems are secured to assure positive steam shut-off and what operator actions, if any, are required.

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

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

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

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

c. Essential portions of the MSSS are protected from the effects of high and moderate energy line breaks and cracks, including pipe whip, jet forces, and environmental effects. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in SRP Section 3.6.

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

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

8. The SAR is reviewed to assure that the applicant has committed to address the potential for steam hammer and relief valve discharge loads, and will take adequate procedures action to minimize such occurrences. Drain pots, line slope and valve operators should be addressed.

IV. EVALUATION FINDINGS

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

The main steam supply system (MSSS) includes all components and piping from the outermost containment isolation valves (for BWRs) [from the steam generator connection (for PWRs)] up to and including the turbine stop valves. The essential portions of the MSSS are designed to quality Group B [for PWRs, from the steam generator to the containment isolation valves, and connected piping up to and including the first valve that is normally closed] [for BWRs, from the outermost containment isolation valves and connecting piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation, but not including the turbine stop and bypass valves]. Those portions of the MSSS necessary to mitigate the consequences of an accident such as a steam line break are designed to the quality standards commensurate with the importance to its safety function, and are designed to the following standards: ________________________. The scope of review of the MSSS for the ____________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system.

The basis for acceptance of the MSSS in our review was conformance of the applicant's design criteria and bases to the Commission's regulations as set forth in the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50. The staff concludes that the plant design is acceptable and meets the requirements of GDC 2, 4, 5, and 34. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," with respect to the ability of structures housing the safety-related portion of the system and the safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods and GDC 4 "Environmental and Missile Design Bases" with respect to structures housing the safety-related portions of the system and the safety-related portions of the system being capable of withstanding the effects of external missiles, and internally-generated missiles, pipe whip and jet impingement forces associated with pipe breaks. The essential portions of the MSSS (as identified in the above discussion) are designed Seismic Category I and housed in a Seismic Category I structure which provides protection from the effects of tornadoes, tornado missiles, turbine missiles, and floods. This meets the positions of Regulatory Guide 1.29, "Seismic Design Classification," Position C.1.a, C.1.e, C.2 and C.3 or C.1.f, C.2 and C.3; Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," Position C.1; and Regulatory Guide 1.117, "Tornado Design Classification," Appendix Positions 2 and 4.

   In addition, the system design capabilities should include the capability to accommodate steam hammer dynamic loads resulting from rapid closure of systems valves (including turbine bypass and stop valves), and safety/relief valve operation without compromising required safety functions. Water entrainment considerations should include provisions for drain pots, line sloping and valve operation. Operating and maintenance procedures are to be reviewed by the applicant to alert plant personnel to the potential for such occurrences and means to minimize such occurrences. This commitment should be stated in the applicants' SAR.

2. The applicant has met the requirements of GDC 5, "Sharing of Structures, Systems, and Components with Respect to the Capability of Shared Systems and Components," important to safety to perform required safety functions. We have reviewed the interconnections from the MSSS of each unit to ___________. The interconnections are designed so that the capability to mitigate the consequences of an accident in either unit and achieve safe shutdown in that unit is retained without reducing the capability of the other unit to achieve safe shutdown.

   or

   Each unit of the ____________ plant has its own MSSS with no interconnections between the safety-related and/or nonsafety- related portions.

3. The applicant has met the requirements of GDC 34, "Residual Heat Removal," with respect to the system function of transferring residual and sensible heat from the reactor system in PWR plants. The MSSS is capable of providing heat sink capacity and pressure relief capability and supplying steam to the steam driven safety-related pumps necessary for safe shutdown. The MSSS is also designed to include the capability to operate the atmospheric pump valves remotely from the control room following a safe shutdown earthquake coincident with the loss of offsite power so that a cold shutdown can be achieved with dependence upon safety-grade components only. This meets the positions in Branch Technical Position RSB 5-1, "Design Requirements of Residual Heat Removal System," and in Issue 1 of NUREG-0138.

V IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implemenation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides, NUREGs and implementation of acceptance criterion subsection II.2, associated with water hammer loads, is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures Systems and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

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

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

7. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

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

9. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1, Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

10. Branch Technical Position RSB 3-1, "Classification of Main Steam Components Other than the Reactor Coolant Pressure Boundary for BWR Plants," attached to SRP Section 3.2.2.

11. Branch Technical Position RSB 3-2, "Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary," attached to SRP Section 3.2.2.

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

13. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, memorandum from Director NRR to NRR Staff."

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.4 CONTROL ROD DRIVE SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

The control rod drive system (CRDS) consists of the control rods and the related mechanical components which provide the means for mechanical movement. General Design Criteria 26 and 27 require that the CRDS provide one of the independent reactivity control systems. The rods and the drive mechanism shall be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences, or under postulated accident conditions. A positive means for inserting the rods shall always be maintained to ensure appropriate margin for malfunction, such as stuck rods. Since the CRDS is a system important to safety and portions of the CRDS are a part of the reactor coolant pressure boundary (RCPB), General Design Criteria 1, 2, 14, and 29 and 10 CFR Part 50,§50.55a, require that the system shall be designed, fabricated, and tested to quality standards commensurate with the safety functions to be performed, so as to assure an extremely high probability of accomplishing the safety functions either in the event of anticipated operational occurrences or in withstanding the effects of postulated accidents and natural phenomena such as earthquakes.

Information in the areas noted below is provided in the applicant's safety analysis report and is reviewed by the MEB in accordance with this SRP section. This information pertains to the CRDS, which is considered to extend to the coupling interface with the reactivity control elements in the reactor pressure vessel. For electromagnetic systems, the review under this SRP section is limited to just the control rod drive mechanism (CRDM) portion of the CRDS. For hydraulic systems, the review covers the CRDM and also the hydraulic control unit, the condensate supply system, and the scram discharge volume. For both types of systems, the CRDM housing should be treated as part of the RCPB; the relevant mechanical engineering information may be presented in this SRP section or by reference to the sections on the RCPB.

If other types of CRDS are proposed or if new features that are not specifically mentioned here are incorporated in CRDS of current types, information should be supplied for the new systems or new features similar to that described below.

1. The descriptive information, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives, is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly.

2. A review is performed of information pertaining to design codes, standards, specifications, and standard practices, as well as to General Design Criteria, regulatory guides, and branch positions that are applied in the design, fabrication, construction, and operation of the CRDS.

   The various criteria, described in general terms above, should be supplied along with the names of the apparatus to which they apply. Pressurized portions of the system which are a part of RCPB are reviewed to determine the extent to which the applicant complies with the Class 1 requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code"). Those portions which are not part of the RCPB are reviewed with other specified parts of Section III, or other sections of the Code. The MEB reviews the non-pressurized portions of the control rod drive system to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. If an experimental testing program is used in lieu of analysis, the program is reviewed to determine whether it adequately covers the areas of concern in stress, deformation, and fatigue.

3. Information is reviewed which pertains to the applicable design loads and their appropriate combinations, to the corresponding design stress limits, and to the corresponding allowable deformations. The deformations are of interest in the present context only in those instances where a failure of movement could be postulated due to excessive deformation and such movement would be necessary for a safety-related function.

   If the applicant selects an experimental testing option in lieu of establishing a set of stress and deformation allowables, a detailed description of the testing program must be provided for review.
   In the preliminary safety analysis report (PSAR), the load combinations, design stress limits and allowable deformations criteria should be provided for review.

   In the final safety analysis report (FSAR), the actual design should be compared with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

   Loadings imposed during normal plant operation and startup and shutdown transients include but are not limited to pressure, deadweight, temperature effects, and anticipated operational occurrences. Loadings associated with specific seismic and other dynamic events are then combined with the above plant-type loads. For BWRs only, the CRDS is reviewed to verify that the system is capable of withstanding adverse dynamic loads such as water hammer. The response to each set of combined loads has a selected stress or deformation limit. The selection of a specific limit is influenced by the probability of the postulated event occurring and the need to assure operation during and after the event.

4. The portion of the SAR is reviewed that describes plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus. For example, the life cycle test program for the CRDS is reviewed. The operability assurance program is reviewed to ascertain coverage of the following:

   a. Life cycle test program.

   b. Proper service environment imposed during test, including appropriate anticipated normal operational occurrences, seismic, and postulated accident conditions.

   c. Mechanism functional tests.

   d. Program results.

In addition, the MEB will coordinate other branches' evaluations that interface with the overall review of the CRDS as follows:

The Core Performance Branch (CPB) will verify fuel system design, including effects of the CRDS on fuel behavior in meeting the requirements of the reactor core design under various normal and accident operating conditions in SRP Section 4.2. The Materials Engineering Branch (MTEB) will review the material aspects of CRDS in SRP Section 4.5.1.

For those areas of review identified above as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

MEB acceptance criteria are based on meeting the requirements of the following regulations:

1. GDC 1 and and 10 CFR Part 50,§50.55a, as its relates to CRDS, requires that the CRDS be designed to quality standard commensurate with the importance of the safety functions to be performed.

2. GDC 2, as it relates to CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

3. GDC 14, as it relates to CRDS, requires that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

4. GDC 26, as it relates to CRDS, requires that the CRDS be one of the independent reactivity control systems which is designed with appropriate

margin to assure its reactivity control function under anticipated normal operation condition.

5. GDC 27, as it relates to CRDS, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of controlling reactivity and cooling the core under postulated accident conditions.

6. GDC 29, as its relates to CRDS, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

Specific criteria necessary to meet the relevant requirements of the regulations identified above are as follows:

1. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of Reference 11.

2. Construction (as defined in NCA-1110 of Section III of the ASME Code, Reference 7) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

   a. Pressurized Portions of Equipment Classified as Quality Group A, B, C (Regulatory Guide 1.26)

      Section III of the ASME Code, Class 1, 2, or 3 as appropriate (Ref. 7).

   b. Pressurized Portions of Equipment Classified as Quality Group D (Regulatory Guide 1.26)

      (1) Section VIII, Division 1 of the ASME Code for vessels and pump casings (Ref. 7).

      (2) Applicable to Piping Systems (American National Standards Institute, ANSI):[1]

         B16.5 Steel Pipe Flanges and Flanged Fittings (Ref. 13).
         B16.9 Steel Butt Welding Fittings (Ref. 14).
         B16.11 Steel Socket Welding Fittings (Ref. 15).
         B16.25 Butt Welding Ends (Ref. 16).
         B31.1 Piping (Ref. 17).
         SP-25 Standards (Ref. 18).
         B16.34 Valves (Ref. 19).

   c. Nonpressurized Equipment (Non-ASME Code)

      Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than those for other plants of

[1]This list can be extended by a staff review and acceptance of other ANSI and MSS standards in the piping system area.

similar design having a period of successful operation. Justification of any decreases should be provided.

3. For the various design and service conditions defined in NB-3113 of Section III of the ASME Code (Ref. 7), load combination sets are as given in Standard Review Plan Section 3.9.3 (Ref. 12). The stress limits applicable to pressurized and nonpressurized portions of the control rod drive systems should be as given in Reference 12 for the response to each loading set. The CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved.

4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and overcoming a stuck rod meet system design requirements.

III. REVIEW PROCEDURES

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

1. The objectives of the review are to determine that design, fabrication, and construction of the control rod drive mechanisms provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

   In the construction permit (CP) review, it should be determined that the design criteria utilize proper load combinations, stress and deformation limits, and that operability assurance is provided by reference to a previously accepted testing program or that a commitment is made to perform a testing program which includes the essential elements listed below. In the operating license (OL) review, the results of any testing program not previously reviewed should be evaluated.

2. The design criteria presented should be evaluated for both the internal pressure-containing portions and other portions of the CRDS. These include the CRDM housing, hydraulic control unit, condensate supply system and scram discharge volume, and portions such as the cylinder, tube, piston, and collect assembly.

   Of particular concern are any new and unique features which have not been used in the past. Pressure-containing components are checked to ensure that they meet the design requirements of the codes and criteria which have been accepted by the Mechanical Engineering Branch, and are identified in Standard Review Plan Section 3.2.2. The review of the functional design of reactivity control systems, including control rod drive systems, is the responsibility of the Reactor Systems Branch (RSB) (see SRP Section 4.6). The loading combinations for the various plant operating conditions are checked for consistency with Reference 12; given these loading combinations, the stress limits of the appropriate code should not be exceeded, or the limits in Reference 12 should not be exceeded if not specified in the listed design code. Exceptions taken by the applicant to any of the accepted codes, standards, or NRC criteria must be identified and the basis clearly justified so that evaluation is possible. Engineering judgment, experience, comparisons with earlier cases and design margins, and consultation with supervisors permit the reviewer to reach a decision on the acceptability of any exceptions posed by the applicant.

The choice of structural materials of construction for the CRDS is reviewed by the MTEB in SRP Section 4.5.1.

3. Loading combinations are defined as those loadings associated with plant operations which are expected to occur one or more times during the lifetime of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, combined with loadings caused by natural or accident events including, for BWRs, water hammer loads. The load combinations which are postulated to occur are specified for each of the design and service conditions as defined in Paragraph NB-3113 of the ASME Code (Ref. 7). These load combinations are defined in Reference 12 and are compared by the reviewer with those provided by the applicant.

   The design stress limits, including fatigue limits, and deformation limits as appropriate to the components of the control rod drive mechanism are compared by the reviewer with those of specified codes, previously designed and successfully operating systems, or with the results of scale model and prototype testing programs.

4. The control rod drive mechanisms of a new design or configuration should be subjected to a life cycle test program to determine the ability of the drives to function during and after normal operating occurrence, seismic, and postulated accident condition over the full range of temperatures, pressures, loadings, and misalignment expected in service. The tests should include functional tests to determine times of rod insertion and withdrawal, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic CRDS, ability to overcome a stuck rod condition, and wear. Rod travel and number of trips expected during the mechanism operational life should be duplicated in the tests.

   The reviewer checks the elements of the test program to be sure all required parameters have been included and finally reviews the test results to determine acceptability. Excessive wear, malfunction of components, operating times beyond determined limits, scram accumulator leakage, etc., all would be cause for retesting.

IV. EVALUATION FINDINGS

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

> The staff concludes that the design of the control rod drive system is acceptable and meets the requirements of General Design Criteria 1, 2, 14, 26, 27, and 29, and 10 CFR Part 50, §50.55a. This conclusion is based on the following:

1. The applicant has met the requirement of GDC 1 and 10 CFR Part 50, §50.55a, with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the control rod drive system are in conformance with the requirements of appropriate ANSI and ASME Codes.

2. The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and anticipated normal operation occurrences with adequate margins to assure its reactivity control function and with extremely low probability of leakage or gross rupture of reactor coolant pressure boundary. The CRDS design capabilities include the ability to accommodate water hammer dynamic loads resulting from rapid opening of the scram insert and withdraw valves and closure of the hydraulic buffer under the worst case loading condition without compromising the safety functions of the system. The specified design transients, design and service loadings, combination of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME Codes and acceptable regulatory positions specified in SRP Section 3.9.3.

3. The applicant has met the requirements of GDC 27 and 29 with respect to designing the control rod drive system to assure its capability of controlling reactivity and cooling the reactor core with appropriate margin, in conjunction with either the emergency core cooling system or the reactor protection system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and implementation of acceptance criterion associated with water hammer loads in BWRs, subsection II.3, is as follows.

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

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

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

4. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

5. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

6. 10 CFR Part 50, Appendix A, General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

7. ASME Boiler and Pressure Vessel Code, Sections III and VIII, American Society of Mechanical Engineers.

8. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

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

10. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

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

12. Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

13. ANSI B 16.5, "Steel Pipe Flanges and Flanged Fittings," American National Standard Institute.

14. ANSI B 16.9, "Wrought Steel Butt Welding Fittings," American National Standard Institute.

15. ANSI B 16.11, "Steel Fittings Steel Welding and Threaded," American National Standard Institute

16. ANSI B 16.25, "Butt Welding Ends - Pipe, Valves, Flanges, and Fittings," American National Standard Institute.

17. ANSI B 31.1, "Power Piping," American National Standard Institute.

18. MSS-SP-25, "Marking for Valves, Fittings, Flanges, and Unions," Manufacturers Standardization Society.

19. ANSI B 16.34, "Steel Valves with Flanged and Butt Welding Ends," American Society of Mechanical Engineers.