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

The review performed under this section pertains to the applicant's program for protecting safety-related components and structures against the effects of postulated pipe breaks both inside and outside containment. The effect that breaks or cracks in high and moderate energy fluid systems would have on adjacent safety-related components or structures has been analyzed with respect to jet impingement, pipe whip, and environmental effects. Several means are used to assure the protection of these safety-related items. They include physical separation, enclosure within suitably designed structures, the use of pipe whip restraints, and the use of equipment shields.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Our review under Standard Review Plan Section 3.6.2 was concerned with the locations chosen by the applicant for postulating piping failures. We also reviewed the size and orientation of these postulated failures and how the applicant calculated the resultant pipe whip and jet impingement loads which might affect nearby safety-related components.

The following discusses several open issues in our review and concludes with our findings which are contingent upon resolution of these open issues.

The response of Question 110.15 is not satisfactory. The applicant should provide a list of all locations where the restraint of one end of a postulated circumferential pipe break was used to reduce the jet force and reaction by reducing the flow area. The applicant should provide justification in each instance that restraint of only one end of a postulated circumferential pipe break would prevent the other end from displacing more than one pipe diameter.

It is stated in the response to Question 110.28 that no mechanistic approaches were used to reduce the flow area of a longitudinal break. The applicant should change the FSAR text to be consistent with this response.

No crack propagation times greater than one millisecond can be assumed unless substantiated by experimental data or analytical theory. The applicant should supply a list of all instances where crack opening times greater than one millisecond were used and supply the needed justification.

All data in Tables 3.6A-1 through 3.6A-13B are due to be updated and completed in April 1980. Our acceptance of Section 3.6.2 is contingent upon final review of these updated tables.

The applicant has stated that after a postulated failure of high energy piping, some non-Category I equipment would be used to bring the plant to a safe shutdown. The applicant must provide assurance that the failure of the seismic Category I piping would not cause failure of the non-Category I equipment.

The applicant states that the use of non-seismic piping inside containment was permitted only on a case-by-case basis with the necessary justification. A list of all non-seismic piping inside containment is requested along with the necessary justification.

The applicant has chosen break and crack locations in non-seismic Category I piping at terminal ends and fittings. We require that such breaks and cracks be chosen at worst-case locations. We consider this an open issue.

We have reviewed the analytical methods used to define the forcing functions at the postulated break or crack locations and jet impingement loadings on

adjacent safety-related structures, systems, and components and find these methods consistent with NRC Standard Review Plan 3.6.2. The applicant references GE Spec. No. 22A2625, GE Report NEDE-10313 and BN-TOP-2, Revision 2, to describe his methods. GE Spec. No. 22A2625 and GE Report NEDE-10313 have been reviewed by NRC staff and found to be acceptable. It should be noted that BN-TOP-2, Rev. 2, has not been accepted. Revision 3 of this report has been accepted; therefore, the FSAR should be clarified to reflect the difference between the two versions.

Subject to resolution of the above open issues, our findings are as follows:

The applicant has proposed criteria for determining the location, type, and effects of postulated pipe breaks in high energy piping systems and postulated pipe cracks in moderate energy piping systems. The applicant has used the effects resulting from these postulated pipe failures to evaluate the design of systems, components, and structures necessary to safely shut the plant down and to mitigate the effects of these postulated piping failures. The applicant has stated that pipe whip restraints, jet impingement barriers, and other such devices will be used to mitigate the effects of these postulated piping failures.

We have reviewed these criteria and have concluded that they provide for a spectrum of postulated pipe breaks and pipe cracks which includes the most likely locations for piping failures, and that the types of breaks and their effects are conservatively assumed. We find that the methods used to design the pipe whip restraints provide adequate assurance that they

will function properly in the event of a postulated piping failure. We further conclude that the use of the applicant's proposed pipe failure criteria in designing the systems, components, and structures necessary to safely shut the plant down and to mitigate the consequences of these postulated piping failures provides reasonable assurance of their ability to perform their safety function following a failure in high or moderate energy piping systems. The applicant's criteria comply with Standard Review Plan Section 3.6.2 and satisfy the applicable portions of General Design Criterion 4.

3.7.3 Seismic Subsystem Analysis

The review performed under Standard Review Plan Section 3.7.3 included the applicant's dynamic analysis of all seismic Category I piping systems. In addition to operating transient loads such as suppression pool loads, this analysis also considers abnormal loadings such as an earthquake.

For the dynamic analysis of seismic Category I piping, each pipe line was idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system was determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next, the mode shapes and the undamped natural frequencies were obtained. The dynamic response of the system was calculated by using the response spectrum method of analysis. For a piping system which was supported at points with different dynamic excitations, the response spectrum analysis was performed using the envelope response spectrum of all support points. Alternately, the multiple excitation analyses methods may have been used where separate acceleration time-histories or response spectra were applied to each piping system support points.

The relative displacement between anchors was determined from the dynamic analysis of the structures. The results of the relative anchor point displacement were used for a static analysis to determine the additional stresses due to relative anchor point displacements.

The applicant's procedures for the dynamic analysis of Category I piping have been reviewed by us and found to be acceptable in part. However, the following open issues must be resolved before we can report our findings.

Standard Review Plan Section 3.7.3, "Seismic Subsystem Analysis," requires five OBE's with a minimum of 10 cycles each to be utilized in fatigue evaluation. This requirement has not been met. The applicant must justify this deviation from Standard Review Plan 3.7.3 or commit to meet our requirements.

Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," outlines the procedures for combining modal responses. Specifically, modes having frequencies falling within 10% of each other are defined as closely spaced modes and must be combined by the absolute sum method. Our review of FSAR Section 3.7.3 cannot be completed until assurance is provided that this criteria has been met or that an equivalent level of safety has been achieved.

Upon resolution of the above open issues, we will report our findings in a supplement to this Safety Evaluation Report.

3.9 Mechanical Systems and Components

The review performed under Standard Review Plan Sections 3.9.1 through 3.9.6 pertains to the structural integrity and operability of various safety-related mechanical components in the plant. Our review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, ventilation ducting, cable trays, and any safety-related piping designed to industry standards other than the ASME Code. We review such issues as load combinations, allowable stresses, methods of analysis, summary results, seismic qualification, preoperational testing, and inservice testing of pumps and valves. Our review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review performed under Standard Review Plan Section 3.9.1 pertains to the design transients, computer programs, experimental stress analyses and elastic-plastic analysis methods that were used in the analysis of seismic Category I ASME Code and non-Code items.

Additionally, we have contracted with Pacific Northwest Laboratories to perform an independent analysis of a sample piping system in the Grand Gulf plant. This analysis will verify that the sample piping system meets the applicable ASME Code requirements, and will also provide a check on the applicant's ability to correctly model and analyze its piping systems. We will report the results of this independent piping analysis in a supplement to this Safety Evaluation Report.

Computer programs were used in the analysis of specific components. A list of the computer programs that were used in the dynamic and static analyses to determine the structural and functional integrity of these components is included in the FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that verification of the computer programs be included. While the required verification is provided for most computer programs, it is lacking for several. The applicant must provide verification for all of the listed computer programs.

It is stated that when ME-632 was verified it did not compute tee stresses per ASME Code, Section III. The applicant should provide assurance that ME-632 now computes tee stresses per ASME Code, Section III, or that the computed tee stresses are conservative with respect to the ASME Code.

Experimental stress analysis was used to verify the design adequacy of piping seismic shock suppressors, pipe whip restraints, and the BWR 6 Orificed Fuel Support. More information on how stresses were determined during the load tests on the BWR 6 Orificed Fuel Support is required. The remainder of the discussion on the experimental stress analysis is adequate.

Elastic-plastic stress analysis methods were used in the evaluation of certain components for the faulted conditions. In general, the information provided on the faulted condition analysis was adequate. More information on the procedures and assumptions used in elastic-plastic analysis of the hydraulic control unit under the SSE faulted condition is requested.

Subject to resolution of these open issues, our findings are as follows:

The methods of analysis that the applicant has employed in the design of all seismic Category I ASME Code Class 1, 2, and 3 components, component supports, reactor internals, and other non-Code items are in conformance with Standard Review Plan Section 3.9.1 and satisfy the applicable portions of General Design Criteria 2, 4, 14, and 15.

The criteria used in defining the applicable transients and the computer codes and analytical methods used in the analyses provide assurance that the calculations of stresses, strains, and displacements for the above noted items conform with the current state-of-the-art and are adequate for the design of these items.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The review performed under Standard Review Plan Section 3.9.2 pertains to the criteria, testing procedures, and dynamic analyses employed by the applicant to assure the structural integrity and operability of piping systems, mechanical equipment, reactor internals and their supports under vibratory loadings. During the present review, the seismic qualification of safety-related mechanical equipment was not considered. This area is within the scope of the Seismic Qualification Review Team (SQRT) which is discussed in Section 3.10 of this Safety Evaluation Report.

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a preoperational testing program. The purpose of these tests is to assure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent.

During the Grand Gulf plant's preoperational and startup test program, the applicant will test various piping systems for abnormal steady-state or transient vibration and for restraint of thermal growth. This test program will comply with the ASME Code, Section III, paragraphs NB-3622, NC-3622, and ND-3622 which require that the designer be responsible, by observation during start-up or initial operation, for ensuring that the vibration of piping systems is within the acceptable levels. In addition, pipe whip restraint initial clearances will be checked, as will snubber response. This test program will consist of a mixture of instrumented measurements and visual observation by qualified personnel.

We have reviewed this test program and asked the applicant in Question 110.33 for more information and clarification. We have the following comments concerning the applicant's response to Question 110.33, which is comprised of eight parts.

(1) The applicant must verify that the types of piping systems included in the test program meets our criteria. We require further information concerning the transient events which could affect these lines.

(4) We do not approve the GE acceptance criteria for steady-state piping vibration or for transient snubber loads. GE stated that the piping stress due to vibration would be maintained below the ASME Code upset limit for primary stress. We believe that the allowable piping stress due to steady-state vibration should be set at some percentage of the material endurance limit. Since the transient events in this test program are expected to occur repeatedly throughout the plant life, we believe that the acceptable snubber load should be the snubber's upset load rating, not its ultimate capacity.

(5) The only listed transient is turbine stop valve closure. The applicant should also include other pump and valve transients.

(6) The applicant has not answered the question, which asked for a list of any BOP piping systems which would be instrumented.

(7) Comparing measured values against calculated values is a generally acceptable approach. The applicant must verify that Bechtel has calculated thermal displacements, steady-state vibration, and transient vibration for all the piping systems for which instrumented measurements will be made.

Also, we will require the applicant to provide a summary of the results of this test program upon its completion.

Based upon our review of FSAR Section 3.9.2 and subject to resolution of the open issues, our findings are as follows:

The 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 is an acceptable program. 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. This test program complies with Standard Review Plan Section 3.9.2 and constitutes an acceptable basis for fulfilling the applicable requirements of General Design Criteria 14 and 15.

During the course of the design and construction of the Grand Gulf plant, the applicant has specified that safety-related mechanical equipment be qualified to be able to perform their safety function during the safe shutdown earthquake and other dynamic events. In those instances where no specific action of the component is required, only the structural integrity of the component need be assured. In many cases, however, the mechanical component must perform a physical movement of some kind to perform its safety function and is necessary for safe shutdown of the plant. These components are termed "active" components. Not only their structural integrity, but also their operability, must be demonstrated The operability of active pumps and valves is discussed in Section 3.9.3 of this Safety Evaluation Report. Our review of the dynamic qualification of mechanical equipment will be handled by our Seismic Qualification Review Team and will be discussed in a supplement to this Safety Evaluation Report. Further discussion of our Seismic Qualification Review Team can be found in Section 3.10 of this report.

Since Grand Gulf is the first of the GE BWR 6 product line, the dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transients must be analyzed. The applicant has outlined his approach for determining the forcing functions considered in the system and component dynamic analyses of reactor structures for normal operation and anticipated transients. These methods are a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design. The forcing function information is combined with dynamic modal analysis to form a

basis for interpretation of the preoperational and initial startup test results. Modal stresses are calculated and relationships are obtained between sensor responses and peak component stresses for each of the lower modes.

The applicant has committed to vibrational measurement and inspection programs to be conducted during preoperational and initial startup testing. Since Grand Gulf is expected to be the first BWR 6, this testing will be in accordance with the guidelines of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," for prototype reactors.

These tests will be conducted in three phases. These are preoperational tests prior to fuel loading, zero-power tests with fuel, and initial startup tests. During preoperational testing, steady-state test conditions will include balanced (two-pump) and unbalanced (one-pump) operation of the recirculation system with flow over the full range up to rated flow. Transient flow conditions will include single and dual pump trips from rated flow. Test duration will ensure that a minimum of 10^6 cycles of vibration will be experienced by the critical components. Inspection of internals will be conducted before and after the test. The zero-power tests with fuel are to verify the anticipated effects of the fuel on the vibration response of internals prior to criticality. Test flow conditions will be similar to the preoperational tests. During the initial startup tests, flow conditions will be similar to the other tests except that power will be up to 100 percent of rated. The primary purpose of these tests is to verify the anticipated effect of two-phase flow.

Vibration sensor types will include strain gages, displacement sensors (linear variable transformers), and accelerometers. Accelerometers will be provided with double integration signal conditioning to give a displacement output. Sensor locations and measured parameters will include the following:

Top of shroud head, lateral acceleration and displacement.
Top of shroud, lateral displacement
Jet pump riser braces, bending and extension strains
Jet pump diffuser, lateral motion or bending strain
Control rod drive housings, bending strain
Incore housings, bending strain
Core spray internal piping, bending strain

We require the applicant to provide a brief summary of the results of this test program upon its completion.

Recently, cracking has been observed in BWR jet pump hold down beams. The resolution of this problem may affect the design or testing of the Grand Gulf jet pumps. Based upon our review of FSAR Sections 3.9.2.3, 3.9.2.4, and 3.9.2.6 and subject to resolution of the above open issues, our findings are as follows:

The preoperational vibration program planned for the reactor internals 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. The conduct of the preoperational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and Standard Review Plan Section 3.9.2, and satisfies the applicable requirements of General Design Criteria 1 and 4.

The applicant has analyzed the reactor, its internals, and unbroken loops of the reactor coolant pressure boundary, including the supports, for the combined loads due to a simultaneous loss-of-coolant accident and safe shutdown earthquake. We cannot complete our review in this area until the applicant submits the information requested in Question 110.29.

Based upon our review of the FSAR Section 3.9.2.5 and subject to resolution of any open items, our findings are as follows:

The dynamic system analysis performed by the applicant provides an acceptable basis for confirming the structural design adequacy of the reactor, its internals, and unbroken piping loops to withstand the combined dynamic loads of postulated loss of coolant accident (LOCA) and the safe shutdown earthquake (SSE). The analysis provides adequate assurance that the

combined stresses and strains in the components of the reactor coolant system and reactor internals do not exceed the allowable 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 system analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity under LOCA and SSE 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. Accomplishment of the dynamic system analysis constitutes an acceptable basis for complying with Standard Review Plan Section 3.9.2 and for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

Our review under Standard Review Plan Section 3.9.3 is concerned with the structural integrity and operability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, or earlier industry standards. This review is divided into four parts, each of which is discussed briefly below.

The first area of review is the subject of load combinations and allowable stresses. Our required loading combinations for the Grand Gulf plant have been included in Question 110.34. With one exception, the applicant has provided a commitment that all ASME Class 1, 2, and 3 components, component supports, core support structures, control rod drive components, and other reactor internals have been analyzed or qualified in accordance with the referenced loading combinations.

This one exception is that the applicant has not included the combined stresses due to SRV and OBE loads in its fatigue calculations. We realize that a position such as the applicant's was accepted during the review of GESSAR. Upon reconsideration, however, we feel that the fatigue contribution attributable to combined SRV and OBE loads should be addressed for those lines whose failure may result in unacceptable consequences such as bypass of the suppression pool. We consider this to be an open issue.

Another open issue related to load combinations is the applicant's method for combining peak responses to multiple dynamic loads. The applicant has used the "square root of the sum of the squares" method (SRSS) for all dynamic responses. Our position, as outlined in NUREG-0484, "Methodology for Combining Dynamic Responses," is that the SRSS method is acceptable for combining peak dynamic responses due to LOCA and SSE for the RCPB. For other dynamic loads and for other ASME Class 1, 2, and 3 components and supports we are currently preparing a generic position which should be available in the near future.

Some of the data in FSAR Table 3.9-2 are missing and the applicant indicates that the data will be supplied in an amendment to the FSAR. We cannot complete our review until this data is available.

We conveyed our position to the applicant describing an acceptable procedure for assuring the early detection of the possible occurrence of cracks in the Grand Gulf feedwater and control rod drive return nozzles and vessel blend radii. Detailed information on the feedwater nozzle and associated sparger design prepared for Grand Gulf is provided in the General Electric Topical Report NEDE-21821, "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report," which was reviewed as part of the generic Task Action Plan A-10, "BWR Nozzle Cracking." For the control rod drive return nozzle the applicant has proposed a generic modification which has been recommended by General Electric and involves capping the nozzle and rerouting the associated flow. This modification is also being reviewed as a part of Task Action Plan A-10.

Our review has concluded that the GE Topical Report NEDE-21821 is an acceptable reference for the Grand Gulf plant. From a mechanical design standpoint we find that the improved interference fit feedwater nozzle spargers used at Grand Gulf are acceptable. However, the required inservice inspection of these nozzles will be discussed elsewhere in this Safety Evaluation Report.

As discussed in our letter to General Electric Company dated January 28, 1980, we have accepted the applicant's proposal to cut and cap the control rod drive return line. More detailed discussions of these issues will be found in the Task Action Plan A-10 final report to be published shortly as NUREG-0619.

Subject to resolution of the above open issues, our findings are as follows:

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. The design and load combinations and associated stress and deformation limits

specified for ASME Code Class 1, 2, and 3 components comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

The second area of our review under Standard Review Plan Section 3.9.3 is the manner in which the applicant has assured the operability of active pumps and valves. Active pumps and valves are those which must perform a mechanical motion in order to shut down the plant or mitigate the consequences of an accident. For instance, under accident conditions, certain valves must open or close, and certain pumps are required to start. On the other hand, inactive pumps and valves are only required to maintain their position during an accident.

We have reviewed the criteria used in the applicant's program for assuring the operability of active pumps and valves. We find these criteria acceptable. We have not yet reviewed how this program was implemented during the applicant's reevaluation of pump and valve operability under combined seismic and suppression pool loads. Our Seismic Qualification Review Team will review the implementation of the applicant's operability assurance program along with the dynamic qualification of the electrical motors, switches, and other appurtenances attached to these active pumps and valves.

Subject to the review by the Seismic Qualification Review Team, our findings are as follows:

The component operability assurance program for ASME Code Class 1, 2, and 3 active valves and pumps provides adequate assurance of the capability of such active components (a) to withstand the imposed design and service loads without loss of structural integrity, and (b) to perform necessary "active" functions (e.g., valve closure or opening, pump operation) during postulated events and conditions expected when plant shutdown is required. The specified component operability assurance test program complies with Standard Review Plan Section 3.9.3 and satisfies the applicable portions of General Design Criteria 1, 2, and 4.

The third area of review in this section concerns the criteria used by the applicant in designing its ASME Class 1, 2, and 3 safety and relief valves, their attached piping, and their supports. We have specifically reviewed the applicant's compliance with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

As noted above, we have not completed our review of the seismic qualification of active valves in general or of safety and relief valves in particular. Also, as noted above, open issues remain in the areas of load combinations and methods for combining dynamic responses.

Subject to resolution of these open issues, our findings are as follows:

The criteria used 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. The criteria used for the design and installation of ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 and are consistent with those specified in Regulatory Guide 1.67 and Standard Review Plan Section 3.9.3.

The fourth area of our review in this section was the criteria used by the applicant in the design of ASME Class 1, 2, and 3 component supports. All component supports have been designed in accordance with Subsection NF of the ASME Code, Section III.

In Question 110.5, we asked the applicant to provide information concerning its design and use of hydraulic snubbers. We require the applicant to supply similar information for its mechanical snubbers.

In its response to Question 110.39, the applicant stated that its BOP mechanical snubbers were submitted to a 100% rated load between 15 and 33 Hz by application of a single load pulse in both tension and compression. We require a better description of this load pulse.

Since some NSSS and BOP snubbers will be subjected to suppression pool related dynamic loads at frequencies greater than 33 hz, the applicant should describe how its snubbers were qualified for these higher frequencies.

We have reviewed the applicant's design criteria pertaining to buckling of component supports and the design of bolts used in component supports. With respect to buckling, we find the applicant's criteria acceptable.

With respect to bolt design, the applicant has supplied information concerning the design of not only the bolts but also the baseplates into which the bolts are inserted and which the bolts connect to the underlying concrete or steel structures. This information has been submitted as a response to our Office of Inspection and Enforcement Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts." The review of this information is being performed jointly by our Office of Inspection and Enforcement and our Office of Nuclear Reactor Regulation. We will report the results of our review in a supplement to this Safety Evaluation Report.

As noted previously, several issues related to load combinations are still open.

Subject to resolution of the above open issues, our findings are as follows:

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 or supported component operability. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

3.9.4 Control Rod Drive Systems

Our review under Standard Review Plan Section 3.9.4 covers the design of the hydraulic control rod drive system up to its interface with the control rods. We reviewed the analyses and tests performed to assure the structural integrity and operability of this system during normal operation and under accident conditions. We also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40-year life.

As previously discussed in Section 3.9.3 of this report, there are open issues related to load combinations.

Subject to resolution of these open issues, our findings are as follows:

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the control rod drive system are in conformance with Standard Review Plan Section 3.9.4. The use of these criteria provide reasonable assurance that the system will function reliably when required and will form an acceptable basis for satisfying the mechanical reliability requirements of General Design Criterion 27.

3.9.5 Reactor Pressure Vessel Internals

Our review under Standard Review Plan Section 3.9.5 is concerned with the load combinations, allowable stress limits, and other criteria used in the design of the Grand Gulf reactor internals. The applicant has stated that the reactor internals have been designed in accordance with Subsection NG, "Core Support Structures," of the ASME Code, Section III. The description of the configuration and general arrangement of the reactor internal structures, components, assemblies, and systems has been reviewed and found to be quite complete. Our review has resulted in the following open issues.

As previously discussed in Section 3.9.3 of this Safety Evaluation Report, there are open issues related to load combinations.

The applicant indicates that when using response spectrum method analysis for the reactor internals, the square root of the sum of the square method will be used for modal combination. Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," outlines acceptable procedures for combining modal responses. Specifically, modes having frequencies falling within 10% of each other are defined as closely spaced modes and should be combined by the absolute sum method. Our review of FSAR Section 3.9.5 cannot be completed until assurance is provided that this criterion has been met or that an equivalent level of safety has been achieved.

The applicant states that the fact that no plastic deformation occurs in the reactor internals components during emergency or faulted conditions demonstrates that no mechanical interferences exist. We do not necessarily agree that not allowing plastic deformations will assure no mechanical interference. It is our position that even elastic deformation must be checked to provide this assurance.

Subject to resolution of these open issues, our findings are as follows:

The specified transients, design and service loadings, and combination of loadings as applied to the design of the Grand Gulf reactor internals provide reasonable assurance that in the event of an earthquake or of a

system transient during normal plant operation, the resulting deflections and associated stresses imposed on these reactor internals would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these reactor internals to withstand the most adverse loading events which have been postulated to occur during the service lifetime without loss of structural integrity or impairment of function. The design procedures and criteria used by the applicant in the design of the Grand Gulf reactor internals comply with Standard Review Plan Section 3.9.5 and constitute an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4, and 10.

3.9.6 Inservice Testing of Pumps and Valves

In Sections 3.9.2 and 3.9.3 of this Safety Evaluation Report we discussed the design and seismic qualification of safety-related pumps and valves in the Grand Gulf facility. The design of these pumps and valves is intended to demonstrate that they will be capable of performing their safety function (open, close, start, etc.) at any time during the plant life. However, to provide added assurance of the reliability of these components, the applicant will periodically test all its safety-related pumps and valves. These tests are performed in general accordance with the rules of Section XI of the ASME Code. These tests verify that these pumps and valves operate successfully when called upon. Additionally, periodic measurements are made of various parameters and compared to baseline measurements in order

to detect long-term degradation of the pump or valve performance. Our review under Standard Review Plan Section 3.9.6 covers the applicant's program for preservice and inservice testing of pumps and valves. We give particular attention to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

The applicant has not yet submitted its program for the preservice and inservice testing of pumps and valves, as requested by Question 110.40; therefore, we have not yet completed our review.

We will report the resolution of this issue in a supplement to this Safety Evaluation Report.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Our review under Standard Review Plan Section 3.10 is concerned with the tests and analyses performed by the applicant to assure the operability of its safety-related instrumentation and electrical equipment in the event of an earthquake or other dynamic event at the Grand Gulf site.

In instances where components have been qualified by testing or analysis to other than current standards (IEEE Std. 344-1975 and Regulatory Guides 1.92 and 1.100), the components will require reevaluation and possible requalification. Our Seismic Qualification Review Team is scheduled to

review and inspect the nuclear steam supply system and balance-of-plant equipment of the Grand Gulf plant. This review will reevaluate the qualification testing and analysis already performed to determine that the effects of multi-axis seismic input and multi-mode equipment response have been properly accounted for. Our review will also determine that the effects of suppression pool hydrodynamic loads were properly considered. On the basis of the review and site visit, the Seismic Qualification Review Team will ascertain whether any nuclear steam supply system or balance-of-plant equipment have to be requalified. Resolution of this issue wil be presented in a supplement to this Safety Evaluation Report.