NUCLEAR ... COMMISSION

JUN 12 1981

Docket Nos. 50-440
and 50-441

Mr. Dalwyn R. Davidson
Vice President - Engineering
The Cleveland Electric Illuminating
Company
P. O. Box 5000
Cleveland, Ohio 44101

Dear Mr. Davidson:

SUBJECT: REQUEST FOR RESOLUTION OF OPEN ISSUES - MECHANICAL ENGINEERING

In the performance of the Perry licensing review, the staff has prepared a draft Safety Evaluation Report (SER) on sections 3.2, 3.6, 3.7 and 3.9 of the FSAR. A copy of this draft is enclosed for your review. Open issues pertaining to the staff review are identified in the draft SER and these will serve as the basis for further discussion and resolution. A list of questions is also enclosed as the appendix to the SER and these serve primarily as clarification to the nature of the open issues.

The staff requests a meeting with your technical staff and consultants from GAI and GE at a mutually agreeable site in approximately 60 days to discuss the open items in the Draft SER. They recommend that this meeting be held in Reading, Pennsylvania during the week of August 3 and they anticipate a meeting of 3 to 5 days. At the conclusion of this meeting, they expect to have all the open issues resolved and to be able to write their final SER for these sections.

In preparation for this meeting, your technical staff will undoubtedly prepare some form of written response to either the open issues in the SER or the questions. The staff would appreciate receiving a draft copy of the written response at least a week before the meeting. Also, since the staff has an outside consultant for this review, they request that a copy of your response be mailed directly to:

Mr. Gordon Beeman
Pacific Northwest Laboratory
P. O. Box 999
Richland, Washington 99352

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After you have reviewed this request, please provide a schedule and place for the meeting. If you require any clarification of this request, please contact M. D. Houston, Project Manager, (301) 492-8593.

Sincerely,

Original signed by

Robert L. Tedesco
Assistant Director for Licensing
Division of Licensing

Enclosure: As stated

cc w/enclosure:
See next page

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MECHANICAL ENGINEERING BRANCH
DRAFT SAFETY EVALUATION REPORT

PERRY NUCLEAR POWER PLANT UNIT I

MECHANICAL ENGINEERING BRANCH
DRAFT SAFETY EVALUATION REPORT
PERRY NUCLEAR POWER PLANT UNIT I

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 Seismic Classification

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of 10 CFR Part 50, Appendix A, in part, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown 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 10 CFR Part 100 guideline exposures. The earthquake for which these plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR Part 100, Appendix A. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components important to safety are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in Regulatory Guide 1.29. Regulatory Guide 1.29, "Seismic Design Classification," is the principal document used in our review for identifying those plant features important to safety which, as a minimum, should be designed to seismic Category I requirements. Our review of the seismic classification of structures, systems, and components (excluding electrical features) of Perry was performed in accordance with the guidance in Standard Review Plan 3.2-1, "Seismic Classification."

The structures, systems, and components important to safety of Perry that are required to be designed to withstand the effects of an SSE and remain functional have been identified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report. Table 3.2-1, in part, identifies major components in fluid systems, mechanical systems, and associated structures designated as seismic Category I. In addition, piping and instrumentation diagrams in the Final Safety Analysis Report identify the interconnecting piping and valves and the boundary limits of each system classified as seismic Category I. We have reviewed Table 3.2-1 and the fluid system piping and instrumentation diagrams and have some question concerning part of this table.

It states in the FSAR that structures, components and systems designated as Safety Class 1, 2, or 3 are classified as seismic Category I except for some portions of the radioactive waste treatment handling and disposal systems. There are several items in Table 3.2-1 that conflict with this statement.

"The seismic classification indicated in Table 3.2-1 meets the requirement of Regulatory Guide 1.29." It is also stated in Section 1.8 that the Perry plant complies with all of the requirements of Regulatory Guide 1.29. Does this mean that seismic Category I cooling water is provided to the recirculation pumps during normal operation and following a LOCA?

What design requirements were used in the design of the reactor pressure vessel skirt and the core support structures?

Quality assurance requirements should be addressed in Table 3.2-1.

The non-seismic classification of the control rods should be justified. Note 7 does not apply to the control rods.

Provide an explanation for the "I, NA" seismic classification for relief valve discharge piping.

How much of the main steam piping, between the M.O. stop valve and the turbine stop valve, is located in the Auxiliary Building?

There appears to be a discrepancy in the seismic classification of the discharge tunnel. The discharge tunnel and the diffusor nozzle are seismic Category I. The tunnel entrance structure and downshaft are not. Provide clarification for this apparent contradiction.

What is the seismic classification of the Containment Vessel Cooling Units?

Note 19 is an exception to Regulatory Guide 1.29 and should be included in Section 1.8.

Based upon our review of FSAR Section 3.2.1 and subject to the satisfactory resolution of the open items, our findings will be as follows.

We have reviewed Table 3.2-1 and the fluid system piping and instrument diagrams and we conclude that the structures, systems, and components important to safety of Perry have been properly classified as seismic Category I items in conformance with Regulatory Guide 1.29, Revision 1.

All other structures, systems, and components that may be required for operation of the facility are not required to be designed to seismic Category I requirements, including those portions of Category I systems such as vent lines, fill lines, drain lines, and test lines on the downstream side of isolation valves and portions of these systems which are not required to perform a safety function.

We conclude that the structures, systems, and components important to safety of Perry that are within the scope of the Mechanical Engineering Branch and are designed to withstand the effects of an SSE and remain functional are properly classified as seismic Category I items in accordance with Regulatory Guide 1.29 and constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2, and is, therefore, acceptable.

3.3.3 System Quality Group Classification

General Design Criterion 1, "Quality Standards and Records," of 10 CFR Part 50, Appendix A requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These fluid system, pressure-retaining components are part of the reactor coolant pressure boundary and other fluid systems important to safety, where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) to retain radioactive material. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is the principal document used in our review for identifying on a functional basis the components of those systems important to safety that are Quality Groups B, C, and D. Section 50.55a of 10 CFR Part 50 identifies those American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Class 1 components that are part of the reactor coolant pressure boundary (RCPB). Conformance of these RCPB components with Section 50.55a of 10 CFR Part 50 is discussed in Section 5.2.1.1 of this Safety Evaluation Report. These RCPB components are designated in Regulatory Guide 1.26 as Quality Group A. Certain other RCPB components which meet the exclusion requirements of footnote 2 of the rule are classified Quality Group B in accordance with Regulatory Guide 1.26. Our review of the quality group classification or pressure-retaining components of fluid systems important to safety for Perry was performed in accordance with the guidance in Standard Review Plan 3.2.2, System Quality Group Classification."

The systems and components important to safety of Perry have been identified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report. Table 3.2-1, in part, identifies the major components in fluid systems such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves and mechanical systems, such as cranes, refueling platforms, and other miscellaneous handling equipment. In addition, the piping and instrumentation diagrams in the Final Safety Analysis Report identify the classification boundaries of the interconnecting piping and valves.

We have reviewed the applicant's use of the NRC Quality Group system in Table 3.2-1 and on the system piping and instrumentation diagrams and we conclude the pressure-retaining components of fluid systems important to safety have been properly classified and meet the guidance in Regulatory Guide 1.26, Revision 2.

We conclude that the applicant's classification of fluid system pressure retaining components important to safety complies with Standard Review Plan Section 3.2.2, Regulatory Guide 1.26 and satisfies the applicable portions of General Design Criterion 1.

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 or 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 design 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 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.

Standard Review Plan 3.6.2 also sets forth certain criteria for the analysis and subsequent in-service inspection of high energy piping in the break exclusion area of containment penetration. Breaks need not be postulated in those portions of piping that meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the additional design requirements outlined in Branch Technical Position MEB 3-1. Additional in-service inspection is also required for those portions of piping.

The following discusses open issues found in our review of FSAR Section 3.6.2. It concludes with our findings contingent upon resolution of all open issues.

In Section 3.6.1 references are made to "elastic/plastic pipe whip restraints or pipe supports which eliminate pipe whip damage." Details of how pipe supports are designed for pipe whip protection and an example of such an analysis are needed.

Pipe whip need only be considered in those high energy piping systems having sufficient capacity to develop a jet stream. The means for determining high and moderate energy lines is found in Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment". This criteria has been used correctly by the applicant. Some additional information is required to clarify this section. How is it determined that the "internal energy level associated with whipping is insufficient to impair the safety function of any system or component to an unacceptable level"? Details should be provided of any flow restrictors used. Methods used to determine fluid reservoirs with sufficient capacity to develop a jet stream should also be provided.

[redacted]

For determining stresses or fatigue usage factors that require a pipe break to be postulated, plant loadings are to be those resulting from normal and upset conditions plus an OBE. Assurances must be provided that loads due to SRV actuation and discharge are included in the upset conditions.

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks due to stress are to be postulated at the following locations:

(1) If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13 must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\, S_m \text{ and Eq. (12)} > 2.4\, S_m$$

or

$$\text{Eq. (10)} > 2.4\, S_m \text{ and Eq. (13)} > 2.4\, S_m$$

(2) Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

Any deviations from the above criteria must be justified.

Are there any high energy Class 2, Class 3 or B31.1 lines? If so, what criteria is used for postulating breaks in these lines?

Any instances with limited break openings or break opening times exceeding one millisecond must be identified. Any analytical methods, representing test results or based on a mechanistic approach, used to justify the above must be provided and explained in detail. This applies to containment and annulus pressurization as well as general pipe break.

For those portions of ASME, Section III, Class 1 piping designed to seismic Category I standards and included in the break exclusion area breaks need not be postulated providing the following stress criteria are met.

(1) If Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not exceed 2.4 S_m, a break need not be postulated.

(2) If Eq. (10) does exceed 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if:

$$\text{Eq. (10)} < 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} < 2.4\ S_m$$

and

$$\text{Eq. (13)} < 2.4\ S_m$$

(3) Breaks need not be postulated as long as the cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, deadweight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all piping in the break exclusion area.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

Are there any Class 2, Class 3, or B31.1 piping in the break exclusion area?. If so, what criteria ~~is~~ are used in their design?

A list of all systems included in the break exclusion areas must be included in the FSAR. In addition, break exclusion areas should be shown on the appropriate piping drawings.

Provide an example of the detailed stress analysis done on a welded attachment to a process pipe. In addition, provide details of the stress analysis done for the head fitting for the main steam line.

When providing protection from pipe whip, assurances must be provided that all potential targets are examined. Provide a definition for limits of strain which are similar to strain levels allowed in restraint plastic members.

"Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any essential system or component." Provide a list of locations where this technique has been used and an example of the studies performed.

[redacted]

When evaluating the effects of jet impingement loads it is the staff's position that all potential targets must be evaluated. Assurances must be provided that your analysis has considered all potential targets. What service limits are used for piping when evaluating jet impingement loads?

Reference is made to the use of a suitable dynamic load factor (DLF). Provide an example of its use. How is it determined that it is suitable?

In the discussion about snubbers, reference is made to "other simultaneous loads". It further states that these loads are combined by SRSS. What are these loads?

"Piping integrity usually does not depend upon the pipe whip restraints for any loading combination." List all those locations and loading combinations where it does. What service limits are used in the design of the pipe whip restraints?

During hot functional testing what critical locations inside containment are monitored?

Standard Review Plan 3.6.2 allows a 10% increase in yield strength to account for strain rate effects. Any locations where an increase in the yield or ultimate strength greater than 10% has been used must be identified. Justification for any increase greater than 10% must also be provided.

Our review of Section 3.6.2 includes all tables and figures. We have several questions pertainint to tables and figures.

Provide a schedule for the completion of any table that is incomplete. Are all postulated break locations in the recirculation system shown (Figure 3.6-66)? Where are breaks postulated in these figures (Figures 3.6-71, 3.6-73, 7.3-74, 3.6-77, 3.6-78, 3.6-79, 3.6-80)? Indicate the location of valves in this line (Figure 3.6-75).

Based on our review of FSAR Section 3.6.2 and subject to the satisfactory resolution of the identified open items, our findings will be 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 support excitation analyses methods may have been used where separate acceleration time-histories were applied to each piping system support points.

The following discusses open issues found in our review of FSAR Section 3.7.3. It concludes with our findings which are contingent upon the resolution of all open issues.

The discussion on "Different Seismic Movement of Interconnected Components" requires some clarification. "The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the SRSS (square root sum of the squares) method." Provide an example of this type of analysis.

What criteria is used to determine whether or not a mode is significant?

"When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacements as obtained above are treated as secondary stresses." Does this statement pertain to piping or supports?

"Seismic analyses were performed for those subsystems that could be modeled to correctly predict the seismic response." What procedure was used for the other systems? Provide an example of those systems and the analysis done.

It is the staff's position that closely spaced modes be combined by one of the procedures identified in Regulatory Guide 1.92. What procedure is used in the BOP design to account for closely spaced modes? What is meant by "Closely spaced phase modes"? Show how modal phasing can be determined from a response spectrum analysis.

Standard Review Plan 3.7.3 requires that 5 OBEs of 10 cycles each be used for design. Any deviations from the requirements of the SRP must be justified. How many OBE cycles are considered in the NSSS and BOP designs?

[redacted]

In the discussion concerning the modeling of piping part (a) discussing decoupling of the main steam and branch lines is not a criteria.

Mention is made of using 33 hertz as a cutoff frequency for seismic analysis. At some point in the FSAR the applicant must address the frequencies of 50 to 60 hertz and greater that come from the suppression pool hydrodynamics.

"For flexible equipment, the equivalent static load is taken as the product of 1.5 times the equipment mass and the peak floor response spectrum value." Regulatory Guide 1.100 allows the use of the 1.5 factor for verifying the integrity of frame type structures. For equipment having configurations other than a frame type structure, justification is required for the use of the 1.5 factor.

When using the double sum method to combine modal responses, the product of the responses of the closely spaced modes should be taken as an absolute value.

Assurances must be provided that the modeling of valves with offset motor operators is detailed enough to provide acceleration values to be used for valve qualification.

"In addition, the effects of modes not included are added to the SRSS response as one term, using the acceleration at the highest frequency from the SRSS response under 33 hertz to obtain the total response." Provide an example of the analysis done here.

The information presented in Table 3.7.11 is not straightforward. Provide an explanation of this table.

Based on our review of FSAR Section 3.7.3 and subject to the satisfactory resolution of the identified open items, our findings will be as follows:

The scope of review of the seismic system and subsystem analysis for the Perry plant included the seismic analysis methods for all Category I systems and components. It included review of procedures used for modeling and evaluating Category I systems and components. The review included design criteria and procedures for evaluation of the interaction of non-Category I piping with Category I piping. The review also included criteria and seismic analysis procedures for reactor internals and Category I piping outside containment.

The system and subsystem 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 and components. 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. A vertical seismic system dynamic analysis is employed for all systems and components.

We conclude that the seismic system and subsystem analysis procedures and criteria proposed by the applicant provide an acceptable basis for the seismic design of systems and components.

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 ot 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, and pre-operational testing. 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 Perry plant. This analysis will verify that the sample piping system meets the applicable ASME Code requirements. 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 used in the dynamic and static analyses to determine the structural and functional integrity of these components must be 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 also be included. The applicant has not provided verification for all of the listed computer programs.

In addition, the programs DYREC and DYNAL are not included in the list of computer programs used.

Any reference to the ASME Boiler and Pressure Vessel Code should include the part being referenced.

How many SRV cycles have been used in the design of components and systems for the NSSS and BOP scope? How many ADS cycles?

It is stated that elastic-plastic methods of analysis may be used for some components. We would like to review the analysis procedures that would be used if an elastic-plastic analysis was done.

Based upon our review of FSAR Section 3.9.1 and contingent on the satisfactory resolution of the open items, our findings will be 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 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. Seismic qualification of safety-related mechanical equipment will be reviewed by the Equipment Qualification Branch.

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 Perry plant's preoperational and startup testing program, the applicant will test various piping systems for abnormal steady-state or transient vibration and for restraint of thermal growth. This test program must 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 startup 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. The test program should consist of a mixture of instrumented measurements and visual observation by qualified personnel. The applicant will be required to provide a summary of the results of this test program upon its completion.

The applicant's discussion of the testing program in the FSAR is too general and should be redone. More detail of what will actually be done must be provided. The applicant has not given a clear description of the NSSS acceptance criteria for steady-state piping vibrations. The BOP program has not been adequately described. What are the acceptance criteria for steady-state vibrations? For transient vibration? Will snubbers be checked? To what transients will the piping be subjected? Which lines, if any, will be instrumented? If not instrumented, how will the visual observations be performed and on what size pipe lines? The staff's position is that acceptance limits for vibration should be based on half the endurance limit as defined by the ASME Code at 10^6 cycles.

In the discussion on thermal expansion testing of the main steam line, reference is made to the piping system shaking down after a few thermal expansion cycles. Provide an explanation of this statement.

It is stated in the FSAR that Perry will be the prototype for the 238 BWR/6. Provide a commitment that the testing program will be equivalent to that required by Regulatory Guide 1.20 for prototype reactors.

"In addition to the above components, vibration measurements of the core spray sparger will be measured during the preoperational testing of that system at the designated prototype 251 BWR/6 plant (Grand Gulf)." Show how this will be applicable to Perry.

It appears that some results from Grand Gulf will be used in the evaluation and qualification of reactor internals at Perry. Show that the similarity between the two sets of internals is sufficient to allow direct comparisons.

"These periods will be determined from a comprehensive dynamic model of the RPV and internals with 12 degrees of freedom." It is not clear what is actually done here. How can a model be comprehensive and have only 12 degrees of freedom?

Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously under-estimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations.

The applicant has described the design of the reactor internals for blowdown loads only. The applicant should also provide information on asymmetric loads. It is, therefore, necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel
b. Core supports and other reactor internals
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.
2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.
3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:
   a. Steam line nozzles to piping terminal ends.
   b. Feedwater nozzle to piping terminal ends.
   c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials* on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

   a. Limited displacement -- break areas
   b. Fluid-structure interaction
   c. Actual time-dependent forcing function
   d. Reactor support stiffness
   e. Break opening times.

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*Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

4. If the results of the assessment on item 3 above indicate loads leading to inelastic action of these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that safety-related components will retain their structural integrity when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

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 pre-operational 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. ~~Testing~~ The applicant should confirm that 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 plants.

These tests will be conducted in three phases. These are preoperational tests prior to fuel loading, zero-power tests with fuel, and initial start-up 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.

The applicant will be required 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 Perry jet pumps. (See IE Bulletin 80-07.)

Based upon our review of FSAR Section 3.9.2.1 and contingent upon the satisfactory resolution of the open items, our findings will be 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 clearnaces 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.

Based upon our review of FSAR Section 3.9.2.3, 3.9.2.4, and 3.9.2.6 and subject to resolution of the above open issue, 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 finalize our review in this area until the applicant submits the information requested under the new loads program. (annulus pressurization)

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 industrial standards. This review is divided into three parts, each of which is discussed briefly below.

The first area of review is the subject of load combinations methodology used in load/response combinations and allowable stresses. 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.

Several references are made throughout this section to allowable stresses for bolting. Specifically, what allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

Are there any Class 1 systems in the BOP scope of responsibility?

The tables in this section provide the major source of information. These tables should be carefully examined by the applicant to ensure clarity and continuity.

Based on our review of FSAR 3.9.3.1 and contingent upon the satisfactory resolution of the open issue, our findings will be 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 [illegible]ce loadings due to potulated 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 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 Protective Devices". We require further clarification of the analyses performed on the SRV piping and supports.

Based upon our review of FSAR section 3.9.3.3 and contingent upon the satisfactory resolution of the open items, our findings will be 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 third area of our review in this section ~~was~~ concerned 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.

We have reviewed the applicant's design criteria pertaining to buckling of component supports. With respect to buckling, we find the applicant's criteria acceptable. As previously discussed, the allowable stress limits for support bolting of Class 1, 2, and 3 components should be provided.

The applicant states that "For the NSSS scope of supply, no valve operators which are mounted on Class 1 piping will be used as component supports.". Are any valve operators mounted on ASME Class 2 and 3 or ANSI B31.1 piping used as component supports? If so, provide a listing of these and an example of the analysis done. Similar information is also required for the BOP scope of responsibility.

Not enough detail is provided on the design and testing of snubbers. Do the design loads on the snubbers include those from SRV discharge and the LOCA? What are the criteria used for the snubber tests? A description of the actual tests are also required.

Into what category are the stresses due to differential anchor support movements placed for supports in the BOP scope of responsibility.

What elastic/plastic analysis has been done on supports? Provide an example of a typical analysis.

Based on our review of FSAR section 3.9.3.4 and contingent upon resolution of the open items, our findings will be 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 by 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.

The applicant has made reference to allowable deformations but they are not defined or listed. This must be included in the FSAR.

Based upon our review of FSAR Section 3.9.4 and contingent upon the satisfactory resolution of the open items, 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 Perry 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.

What feedwater sparger design is used at Perry? The applicant should provide a commitment to NUREG-0619.

Have the reactor internals placed in the "other internals" category been seismically analyzed to show that they will not compromise the integrity of seismically qualified reactor internals during the SSE.

Based upon our review of FSAR Section 3.9.5 and contingent upon the satisfactory resolution of the open items, our findings will be as follows:

The specified transients, design and service loadings, and combination of loadings as applied to the design of the Perry 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 Perry 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 of safety-related pumps and valves in the Perry 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 applicants 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 must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the revised rules of 10 CFR, Part 50, Section 50.55a, paragraph (g).

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

Any requests for relief from ASME Section XI should be submitted as soon as possible.

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems, thus causing the inner-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shut-down or system siolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolat n point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

We will report the resolution of the e issues in a supplement to the Safety Evaluation Report.

APPENDIX

QUESTIONS ON PERRY FSAR

QUESTIONS ON PERRY FSAR

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

3.2.1, Page 3.2-1

It states in the FSAR that structures, components and systems designated as Safety Class 1, 2, or 3 are classified as Seismic Category I except for some portions of the radioactive waste treatment handling and disposal systems. There are several items in Table 3.2-1 in conflict with this statement.

3.2.1, Page 3.2-2

"The seismic classification indicated in Table 3.2-1 meets the requirements of Regulatory Guide 1.29." It is also stated in Section 1.8 that the Perry plant complies with all the requirements of Regulatory Guide 1.29. Does this mean that seismic Category I cooling water is provided to the recirculation pump during normal operation and following LOCA?

Table 3.2-1, Page 3.2-9

Quality assurance requirements should be addressed in this table.

Table 3.2-1, Page 3.2-9

What design requirements were used in the design of the reactor pressure vessel skirt?

Table 3.2-1, Page 3.2-9

Justify the non-seismic classification of the control rods. Note 7 does not apply to the control rods.

Table 3.2-1, Page 3.2-9

Provide an explanation for the "I, NA" seismic classification for relief valve discharge piping.

Table 3.2-1, Page 3.2-10

How much of the main steam piping, between the M.O. stop valve and the turbine stop valve, is located in the Auxiliary Building?

Table 3.2-1, Page 3.2-24

There appears to be a discrepancy in the seismic classification of the discharge tunnel. The discharge tunnel and the diffusor nozzle are seismic Category I. The tunnel entrance structure and downshaft are not. Provide clarificatioo for this apparent contradiction.

Table 3.2-1, Page 3.2-25

What is the seismic classification of the Containment Vessel Cooling Units?

Table 3.2-1, Page 3.2-34

Note 19 is an exception to Regulatory Guide 1.29 and should be included in Section 1.8.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

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

3.6.2, Page 3.6-7

In Section 3.6.1 references are made to "elastic/plastic pipe whip restraints or pipe supports which eliminate pipe whip damage". Details of how pipe supports are designed for pipe whip protection and an example of such an analysis are needed.

3.6.2.1.4, Page 3.6-10

How is it determined that "The internal energy associated with whipping is insufficient to impair the safety function of any structure, system or component to an unacceptable level"?

[redacted]

3.6.2.1.5, Page 3.6-11

Plant loading conditions for evaluating pipe break are to include normal and upset conditions plus an OBE. Assurance must be provided that SRV discharge loads are included in the upset conditions.

3.6.2.1.5, Page 3.6-11

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks due to stress are to be postulated at the following locations:

(1) If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} > 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (13)} > 2.4\ S_m$$

(2) Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

Any deviations from the above criteria must be justified.

3.6.2.1.5, Page 3.6-11

Are there any high energy Class 2, Class 3, or B31.1 lines? If so, what criteria is used for postulating breaks in these lines?

3.6.2.1.6, Page 3.6-13

Any instances where longitudinal break areas are less than one circumferential pipe area must be identified. The analytical methods representing test results and based on a mechanistic approach must be explained or justified. Provide examples of a typical analysis.

3.6.2.1.6, Page 3.6-14

How are energy reservoirs of sufficient capacity to develop a jet flow determined? What are justifiable line restrictions? Provide the justification. Any instances where flow limiters are used should be identified and justified.

3.6.2.1.7.1, Page 3.6-15

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks need not be postulated providing the following stress criteria is met.

(1) If Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not esceed 2.4 S_m, a break need not be postulated.

(2) If Eq. (10) does exceed 2.4 S_m, Then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if

$$\text{Eq. (10)} < 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} < 2.4\ S_m \text{ and Eq. (13)} < 2.4\ S_m$$

(3) Breaks need not be postulated as long as the cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, dead weight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all piping in the break exclusion area.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

3.6.2.1.7.1, Page 3.6-15

Are there any Class 2, Class 3 or B31.1 piping in the break exclusion areas? If so, what criteria is used for their design?

3.6.2.1.7.1, Page 3.6-15

A list of all systems in the break exclusion area is needed. Break exclusion area should be shown on the appropriate piping drawings.

3.6.2.1.7.2, Page 3.6-15

Provide an example of the detailed stress analysis done on a welded attachment to the process pipe. In addition, provide details of the stress analysis done on the head fitting for the main steam line.

3.6.2.2.1, Page 3.6-17

Provide a list of all locations where limited break opening areas have been used. Provide justification for each location and details of any inelastic analysis used.

3.6.2.2.1, Page 3.6-17

Provide a list of all locations where break opening times greater than one millisecond have been used. Provide and justify any experimental data and analytical theory.

3.6.2.2.2, Page 3.6-20

Provide assurance that all potential targets are evaluated when considering pipe whip.

3.6.2.2.2, Page 3.6-20

Provide a definition for limits of strain which are similar to strain levels allowed in restraint plastic member.

3.6.2.2.2, Page 3.6-20

"Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any essential system or component." Provide a list of where this technique has been used and an example of the studies performed.

[redacted]

3.6.2.3.1, Page 3.6-23

It is the staff's position that when evaluating jet impingement loads all potential targets must be evaluated. Provide assurances that your analysis for jet impingement affects have included all possible targets.

3.6.2.3.1, Page 3.6-29

What service limits are used for piping when evaluating jet impingement loads?

3.6.2.3.1, Page 3.6-30

How is it determined that the dynamic load factor (DLF) is suitable? Provide an example of its use.

3.6.2.3.1, Page 3.6-30

For snubbers, what are the "other simultaneous loads" that are combined by the SRSS method?

3.6.2.3.3, Page 3.6-33

"Piping integrity usually does not depend upon the pipe whip restraints for any loading combination." List all those locations where integrity of the piping depends upon the pipe whip restraints.

3.6.2.3.3, Page 3.6-33

What service limits are used in the design of the pipe whip restraints?

3.6.2.3.3.1, Page 3.6-33

What critical locations inside containment are monitored during hot functional testing?

3.6.2.3.3.1, Page 3.6-40

Any locations where the increase in the yield or ultimate strengths, of the material used for pipe whip restraints, exceeds 10% must be identified. Justification for any increase greater than 10% must also be provided.

3.6.2, Tables

Provide a schedule for the completion of any table that is incomplete.

3.6.2, Figure 3.6-66

Are all postulated break locations in the recirculation system shown?

3.6.2, Figures 3.6-71, 3.6-73, 7.3-74, 3.6-77, 3.6-78, 3.6-79, 3.6-80

Where are breaks postulated in these figures?

3.6.2, Figure 3.6-75

Indicate the location of valves in this line.

3.7.3 Seismic Subsystem Analysis

3.7.2.1.2.5, Page 3.7-11

The discussion on "Different Seismic Movement of Interconnected Components" requires some clarification. "The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the SRSS (square root sum of the squares) method." Provide an example of what was done here.

3.7.2.1.2.5, Page 3.7-11

What criteria was used to determine whether or not a mode was significant?

3.7.2.1.2.5, Page 3.7-11

"When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses." Does this statement pertain to piping or supports?

3.7.3.1.1, Page 3.7-20

"Seismic analyses were performed for those subsystems that could be modeled to correctly predict the seismic response." What procedure was used for the other systems? Provide an example of some of those systems.

3.7.3.1.1, Page 3.7-21

What is meant by "Closely spaced in phase modes"?

3.7.3.2.1, Page 3.7-21

How many stress cycles are used in the BOP design?

[redacted]

3.7.3.3.2.1, Page 3.7-23

Part (a) discussing decoupling of main steam and branch lines is not a criteria.

3.7.3.3.2.2, Page 3.7-24

Mention is made of using 33 hertz as a frequency cutoff for seismic analysis. At some point in the FSAR the applicant must address the frequencies of 50 to 60 hertz and greater than come from the suppression pool hydrodynamics.

3.7.3.5, Page 3.7-25

"For flexible equipment, the equivalent static load is taken as the product of 1.5 times the equipment mass and the peak floor response spectrum value." Regulatory Guide 1.100 allows the use of the 1.5 factor for verifying the integrity of frame type structures. For equipment having configurations other than a frame type structure, justification is required for use of the 1.5 factor.

3.7.3.7.1, Page 3.7-26

What procedure is used for combining closely spaced modes of systems in the BOP scope?

3.7.3.7.2, Page 3.7-26

The referenced equation should be as follows

$$R = \left[\sum_{K=1}^{N} \sum_{S=1}^{N} \left| R_K R_S \right| \varepsilon_{KS} \right]^{1/2}$$

3.7.3.8.1, Page 3.7-28

Justification must be provided that the modeling of valves with offset motor operators is detailed enough to provide acceleration values to be used for valve qualification.

3.7.3.8.1, Page 3.7-28

"In addition, the effects of the modes not included are added to the SRSS response as one term, using the acceleration at the highest frequency from the SRSS response under 33 hertz to obtain the total response." Provide an example of what was done here.

Table 3.7-11, Page 3.7-54

Provide a detailed explanation of the information in this table.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9, Page 3.9-1

Any references to the ASME Boiler and Pressure Vessel Code should indicate what part is being referenced.

3.9.1.2, Page 3.9-1

Methods of verification are required for all NSSS computer codes used in the analysis.

3.9.1.2.6, Page 3.9-16

All computer programs used in the design and analysis of systems and components within the BOP scope must be listed. Methods of verification are required for all BOP programs.

3.9.1.4.12, Page 3.9-26

It is stated that elastic-plastic methods of analysis may be used for some components. We would like to review the analysis procedures that would be used if an elastic-plastic analysis was done.

3.9.2, Page 3.9-27

More detail is needed for the NSSS and BOP preoperational vibration testing program. What locations will be monitored. What types of instrumentation will be used. What are the actual values that will be used for deflection and stress limits.

The staff's position is that acceptance limits for vibration should be based on half of the endurance limit as defined by the ASME Code at 10^5 cycles. We will require a copy of your results from your preoperational vibration testing program.

3.9.2.1.2, Page 3.9-29

"The piping system does 'shakedown' after a few thermal expansion cycles." Provide an explanation of this statement.

3.9.2.4, Page 3.9-65

"In addition to the above components, vibration measurements of the core spray sparger will be measured during preoperational testing of that system at the designated prototype 251 BWR/6 plant (Grand Gulf)." Show how this is applicable to Perry.

3.9.2.4.1, Page 3.9-66

Provide a commitment that Perry will be in compliance with Regulatory Guide 1.20 for prototype reactors.

3.9.2.5, Page 3.9-67

"These periods will be determined from a comprehensive dynamic model of the RPV and internals with 12 degrees of freedom." It is not clear what is actually done here. How can a model be comprehensive and have only 12 degrees of freedom?

3.9.2.6, Page 3.9-68

It appears that some results from Grand Gulf will be used in the evaluation and qualification of the reactor internals at Perry. Show that the similarity between the two sets of internals is sufficient to allow direct comparisons.

3.9.3, Page 3.9-68

Several references are made throughout this section to allowable stresses for bolting. Specifically, what allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

3.9.3.1.2, Page 3.9-78

Are there any Class 1 systems in the BOP scope of responsibility?

3.9.3.4.1, Page 3.9-107

"For the NSSS scope of supply, all valve operators which are mounted on Class 1 piping will not be used as attachment points for component supports." What about Class 2 and 3 piping? This question also applies to the BOP scope of responsibility.

3.9.3.4.1, Page 3.9-109

Provide more detail on the testing done on snubbers.

3.9.3.4.4, Page 3.9-112

What elastic-plastic analysis has been done on supports? Provide an example of this analysis.

3.9.4.3, Page 3.9-114

Reference is made to allowable deformation in the title of this section but there is no discussion of allowable deformations in the text.

3.9.5.1.1.8, Page 3.9-120

Recently, cracking has been observed in BWR jet pump holddown beams. The resolution of this problem may affect the design or testing of the Perry jet pumps (see I&E Bulletin 80-07).

3.9.5.1.1.10, Page 3.9-121

What feedwater sparger design and Control Rod Drive Return Line modifications is used at Perry? Provide a commitment to NUREG-0619.

3.9.5.3.3, Page 3.9-129

Have the reactor internals placed in the "other internals" category been seismically analyzed to show that they will not compromise the integrity of seismically qualified reactor internals?

3.9.6, Page 3.9-131

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to pretect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak-tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specification which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

Table 3.9-1, Page 3.9-134

Does this table apply to Perry?

Table 3.9-1, Page 3.9-135

What does "1*****" refer to?

Table 3.9-1, Page 3.9-135

How many ADS cycles are included in the design of Perry?

Table 3.9-1, Page 3.9-136

Standard Review Plan 3.9 requires 5 OBEs of 10 cycles each. If fewer cycles are used, justification must be provided.

Table 3.9-3, Page 3.9-141

The acceptance criteria should reference the ASME Code Service Limits. A similar table is needed for the BOP.

Table 3.9-3a, Page 3.9-143

"The results of stress and fatigue usage analysis are given in detail in the vessel manufacturer's stress report and in new loads evaluation by GE within the code limits." Provide clarification of this statement.

Table 3.9.3m, 3.9.3o, 3.9.3q and 3.9.3h

Some values in these tables are missing. Provide a schedule for their completion.

Table 3.9-3s, Page 3.9-225

Provide an explanation for the results in this table.

Table 3.9-28, Page 3.9-292

Where are the loads used in this table defined? How are these loads combined?

Table 3.9-32, Page 3.9-297

Has Eq. b) been used? If so, provide the supporting data. If not, delete the equation from the table.

Table 3.9-33, Page 3.9-298

Have Eqs. e), f), or g) been used? If so, provide the supporting data. If not, delete these equations from the table.

Table 3.9.34, Page 3.9-301

Has Eq. c) been used. If so, provide the supporting data. If not, delete the equation from the table.

ADDITIONAL QUESTIONS

Table 3.2-1, Page 3.2-9

What design requirements were used in the design of the core support structures?

3.6.2.1.6, Page 3.6-13

Regardless of the ratio of longitudinal to hoop stress, both a longitudinal split and a circumferential break should be postulated at any location where the cumulative usage factor is greater than 0.1.

3.9.1.1.1, Page 3.9-1

How many cycles due to SRV discharge are included in the analysis?

3.9.2.5, Page 3.9-67

Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations.

The applicant has described the design of the reactor internals for blowdown loads only. The applicant should also provide information on asymmetric loads. It is, therefore, necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel
b. Core supports and other reactor internals
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.
2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.
3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:
   a. Steam line nozzles to piping terminal ends.
   b. Feedwater nozzle to piping terminal ends.
   c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials* on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other

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*Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

a. Limited displacement -- break areas
b. Fluid-structure interaction
c. Actual time-dependent forcing function
d. Reactor support stiffness
e. Break opening times.

4. If the results of the assessment on item 3 above indicate loads leading to inelastic action of these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that safety-related components will retain their structural integrity when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

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 pre-operational 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.

3.9.3.3-2, page 3.9-106

Provide justification for using a modified static analysis ~~analysis~~ on the safety relief valve piping in the suppression pool and explain what is used for the "conservative dynamic load factor" in the analysis.

Provide the time-history transient forces resulting from the SRV actuation used in the SRV piping and support design including the loads developed from the discharging water slug.

Discuss the types of supports used on the SRV piping in both the drywell and suppression pool and provide drawings of the supports.

Provide the type of safety relief valves used in the plant , the valve opening time, and the sequences of valve actuation used in the analysis.

3.9.3.4.6, page 3.9-113

Are the stress due to differential anchor movements considered as primary or secondary stresses for BOP supports?

TO ALL APPLICANTS:

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9 This examination should be made after snubber installatio^ but not more than six months prior to initial system pre-operational testing, .nd should as a mimimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifictions.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins. fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1,4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

110.0 MECHANICAL ENGINEERING BRANCH

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

The essential instrumentation lines to be inspected should include (but are not limited to) the following:

a) Reactor pressure vessel level indicator instrumentation lines (used for monitoring both steam and water levels).

b) Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow).

c) Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation).

d) Control rod drive lines inside containment (not normally pressurized but required for scram).

[redacted] MECHANICAL ENGINEERING BRANCH

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