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Docket Nos. 50-458/459

Mr. William Cahill, Jr.
Senior Vice President
River Bend Nuclear Group
Gulf States Utilities Company
P. O. Box 2951
Beaumont, Texas 77704
Attn: Mr. J. E. Booker

Dear Mr. Cahill:

Subject; Early Trans mittal of Mechanical Engineering Draft SER Evaluation/ Questions for the River Bend Station (Units 1 and 2)

The Draft SER for River Bend is scheduled for issuance in early April, 1982. However, the NRC mechanical engineering review staff desires to convene a meeting with GSU, S&W and the NSSS in mid-June, 1982 to discuss and resolve any and all open items/issues in preparing for the final SER, which is scheduled for release on October 4, 1982. It has therefore been decided to provide you with an advance copy of their draft SER evaluation and related questions, which are enclosed, to ensure that you may have sufficient time to prepare for the meeting in June.

It is accordingly requested that arrangements be made to schedule the meeting, preferably at Stone & Webster, on/or about June 14, 1982. A three to five day meeting is envisioned and the meeting agenda structured as such. We expect that the meeting participants will be prepared to resolve the open items/issues, and to commit a response date for those areas which cannot be satisfactorily resolved, so that we may accurately address them in the final SER. The meeting agenda should be submitted not later than May 15, 1982 to allow us to issue the formal meeting notice sufficiently in advance of the meeting.

If there are any questions pertaining to this request or on the enclosed evaluation/questions contact either R. Perch or John Stefano of my staff. Your immediate attention to this request will be most appreciated and is urged.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

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OFFICE	Enclosure:		DL:LB#2/PM	DL:LB#2/BC			
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DATE	cc: See	next page	3/12/82	3/12/82			

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3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

3.2.1 Seismic Classification

The staff has reviewed the material submitted in the FSAR for Section 3.2.1 and finds it to be generally acceptable. A few items requiring clarification and further justification have been identified dealing with the classification of certain systems associated with the Diesel Generators. These open items have been transmitted to the applicant. Upon resolution of these open items, our findings will be as follows:

Structures, systems and components (excluding electrical features) that are important to safety and that are required to withstand the effects of a safe shutdown earthquake and remain functional have been classified as seismic Category I items and have been identified in an acceptable manner in Tables 3.2-.1 and on system piping and instrumentation diagrams in the SAR. Other structures, systems and components that may be required for operation of the facility (excluding electrical features) need not be designed to seismic Cateogry I requirements. The structures, systems and components not required to be designed to seismic Category I include those portions of Category I systems such as vent lines, drain lines, fill lines and test lines on the downstream side of isolation valves and those portions of the systems which are not required to perform a safety function.

The staff concludes that the structures, systems and components important to safety that are within the scope of the Mechanical Engineering Branch have been properly classified as seismic Category I items and meet the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena" and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." This conclusion is based on the applicant having met the requirements of General Design Criterion 2, and 10 CFR Part 100, Appendix A, by having properly classified their structures, systems and components (SSC) important to safety as seismic Category I items in accordance with the positions of Regulatory Guide 1.29, "Seismic Design Classification" and by our conclusion that the identified SSC are the plant features 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, and (3) the capability to prevent and mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

3.2.2 System Quality Group Classification

The staff has reviewed the material presented in FSAR Section 3.2.2 and find it generally acceptable. The applicant has taken exception to certain sections of Regulatory Guide 1.26. While some of these exceptions are acceptable, others require further clarification and justification before they can be accepted. These exceptions have been addressed as questions transmitted to the applicant. Upon resolution of these open issues, our findings will be as follows:

Pressure-retaining components of fluid systems important to safety such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves have been classified Quality Group A, B, C, or D and have been identified in an acceptable manner in Table 3.2-3 and on system piping and instrumentation diagrams in the SAR. These components have been constructed to quality standards commensurate with the importance of the safety function to be performed. The review of Quality Group A and B (ASME Section III, Class 1 and 2) reactor coolant pressure boundary components is discussed in Section 5.2.1.1 of the SER. Other Quality Group B components of systems identified in Position C.1.a through C.1.e of Regulatory Guide 1.26 are constructed to ASME Section III, Class 2.

Components in systems identified in Position C.2.a through C.2.d of Regulatory Guide 1.26 are constructed to Quality Group C standards, ASME Section III, Class 3. Components in systems identified in Position C.3 of Regulatory Guide 1.26 are constructed to Quality Group D standards such as ASME Section VIII and ANSI B31.1.

The staff concludes that pressure-retaining components of fluid systems important to safety have been properly classified as Quality Group A, B, C, or D items and meet the requirements of General Design Criterion 1, "Quality Standards and Records". This conclusion is based on the applicant having met the requirements of General Design Criterion 1 by having properly classified these pressure-retaining components important to safety Quality Group A, B, C, or D in accordance with the positions of Regulatory Guide 1.26, "Quality Group Classifications and Standards", and by our conclusion that the identified pressure-retaining components are those necessary (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 contain radioactive materials.

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

The staff has reviewed the material submitted in the FSAR for Section 3.6.2 and finds that it in general covers all topics requiring discussion. However, several areas need further clarification and justification and inconsistencies are present in a few areas. Additional justification and clarification are required for the limits used in the criteria for the no break zone, the pipe break criteria, the pipe thrust coefficients, jet impingement analyses and pipe whip restraint design. In addition, a very large amount of information will not be available until completion of the new loads program. Our review cannot be completed without this information. These open items have been transmitted to the applicant. Upon resolution of these open items, our finding will be as follows:

The staff evaluation concludes that the pipe rupture postulation and the associated effects are adequately considered in the plant design, and therefore are acceptable and meet the requirements of General Design Criterion 4. This conclusion is based on the following:

1. The proposed pipe rupture locations have been adequately assumed and the design of piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingement provide adequate protection to the integrity and functionality of safety-related structures, systems, and components.

2. The provisions for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid provide adequate assurance that design basis loss-of-coolant accidents will not be aggravated by sequential failures of safety-related piping, and emergency core cooling system performance will not be degraded by these dynamic effects.

3. The proposed piping and restraint arrangement and applicable design considerations for high- and moderateenergy fluid systems inside and outside of containment, including the reactor coolant pressure boundary, will provide adequate assurance that the structures, systems and components important to safety that are in close proximity to the postulated pipe rupture will be protected. The design will be of a nature to mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and

maintained in a safe shutdown condition in the event of a postulated rupture of a high or moderate energy piping system inside or outside of containment.

3.9 MECHANICAL SYSTEMS AND COMPOSITES

3.9.1 Special Topics for Mechanical Components

The staff has reviewed the material submitted in the FSAR for Section 3.9.1 and find it generally acceptable. However, more information is required on the verification of certain computer programs. There are also questions regarding the completeness of the design transients and the number of cycles. Additional clarification is needed regarding tests performed on a limited number of components. These open items have been transmitted to the applicant. Upon resolution of these open items, our findings will be as follows:

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

1. The applicant has met the relevant requirements of General Design Criteria 14 and 15 by demonstrating that the design transients and resulting loads and combinations with appropriate specified design and

service limits which the applicant has used for designing Code Class 1 and CS components and supports, and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

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

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

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

The staff has reviewed the material for Section 3.9.2 provided in the FSAR and find it generally acceptable. However, the number of OBE stress cycles considered in the NSSS scope is not in compliance with Standard Review Plan. This is an open item. Additional information regarding the preoperational test program is required. In particular, more information is required on the locations of measurements and visual inspections, the limits for steady state and transient vibration and the limits and tolerances for thermal expansion. Compliance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", is required. These open items have been transmitted to the applicant and upon their resolution, the staff's finding will be as follows:

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

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

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

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

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

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

assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events.

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

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

The staff has reviewed the information in the FSAR for Section 3.9.3 and find it to be generally acceptable except in the areas of functional capability and design limits. Compliance with NUREG-0800 must be shown. In addition, further clarification is required on certain aspects of support design. These open issues have been transmitted to the applicant and upon resolution, the staff's finding shall be as follows:

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

Code Class 1, 2, and 3 pressure retaining components in systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

2. The applicant has met the requirements of 10 CFR Part 50, §50.55a and General Design Criteria 1, 2, and 4 with respect to the criteria used for design and installation of ASME Code Class 1, 2, and 3 overpressure relief devices by insuring that safety and relief valves and their installations are designed to standards which are commensurate with their safety functions, and that they can accommodate the effects of discharge due to normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from the safe shutdown earthquake. The relevant requirements of General Design Criteria 14 and

15 are also met with respect to assuring that the reactor coolant pressure boundary design limits for normal operation including anticipated operational occurrences are not exceeded. The criteria used by the applicant in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

3. The applicant has met the requirements of 10 CFR Part 50, §50.55a and General Design Criteria 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports by ensuring that component supports important to safety are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as

postulated events such as loss-of-coolant accidents and the dynamic effects resulting from the safe shutdown earthquake. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, has met the positions and criteria of Regulatory Guides 1.124 and 1.130 and are in accordance with NUREG-0484 and NUREG-0609. The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity.

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

3.9.4 Control Rod Drive Systems

The staff has reviewed the information in the FSAR for Section 3.9.4 and find it generally acceptable. If the prototype information is not available from the Kuo Sheng I in a timely manner, the applicant may have to make alternate plans. This concern has been transmitted to the applicant. Upon receiving acceptable prototype information, the staff's findings will be as follows:

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

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

2. The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and anticipated normal operation occurrences with adequate margins to assure its reactivity control function and with extremely low probability of leakage or gross rupture of reactor coolant pressure boundary. The specified design transients, design and service loadings, combination of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME Codes and acceptable regulatory positions specified in SRP Section 3.9.3.

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

3.9.5 Reactor Pressure Vessel Internals

The staff has reviewed the material in the FSAR for Section 3.9.5. This material is generally acceptable except that additional clarification and justification are required for the stress, deformation and buckling limits given. This open issue has been transmitted to the applicant. Upon resolution of the open issue, the staff's findings will be as follows:

The staff concludes that the design of reactor internals is acceptable and meets the requirements of General Design Criteria 1, 2, 4, and 10 and 10 CFR Part 50, §50.55a. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 1 and 10 CFR Part 50, §50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.

2. The applicant has met the requirements of GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated loss-of-coolant accidents with sufficient margin to assure that capability to perform its safety functions is maintained and the specified acceptable fuel design limits are not exceeded.

The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components 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 structures and components 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 structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3.9.6 Inservice Testing of Pumps and Valves

The staff has reviewed the information in the FSAR for Section 3.9.6 and find it incomplete. The specifics of the inservice testing program for pumps and valves are lacking. The criteria upon which the program will be based needs clarification and amplification. These open items have been transmitted to the applicant. Upon resolution of these open issues, the staff's findings will be as follows:

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

RIVER BEND QUESTIONS

Section 3.2.1

Table 3.2-1

It is the staff's position that certain systems important not identified in Regulatory Guide 1.26 should be classified Quality Group C, or its equivalent. Among these systems are: diesel fuel oil storage and transfer system, diesel engine cooling water system, diesel engine lubrication system, diesel engine starting system, and diesel engine combustion air intake and exhaust system. Justify the absence of a quality group classification of portions of those systems listed below:

Diesel Generator Cooling Water System

Diesel Generator Starting System

Diesel Generator Lubrication System

Diesel Generator Combustion Air Intake and Exhaust System

Section 3.2.2

Page 3.2-6, Section 3.2.2.2.3

Exceptions to Regulatory Guide 1.26; Sections C.1.e and C.2.e needs further clarification and justification. ~~Please provide information to support the contention that the~~

~~safety/relief valve is acceptable~~. ~~Also~~ provide a list of all lines to which these exceptions are intended to apply.

Section 3.6.2.1.5.2.1A.2.a

The design stress and fatigue limits for Class I piping in the containment penetration areas are not in compliance with Standard Review Plan 3.6.2 and BTP MEB 3-1. If the maximum stress range of Equation (10) exceeds 2.4 Sm, both Equations (12) and (13) must be less than 2.4 Sm. In all cases the cumulative usage factor must be less than 0.1.

~~Page 3.6A-16~~

~~Section 3.6.2.1.5.2.1A.2.a~~

~~Are occasional loads considered in the evaluation of Equations (9) and (10)?~~

Page 3.6A-16

Section 3.6.2.1.5.2.1A.2.b

The maximum stress range as calculated by the sum of Equation (9) and (10) should consider sustained loads, occasional loads and thermal expansion. Have occasional loads been included as per BTP MEB 3-1?

Page 3.6A-17

Section 3.6.2.1.5.2.1A.2.g

Provide assurance that the 100% volumetric inservice examination of all pipe welds will be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

~~Page 3.6A-18~~

~~Section 3.6.2.1.5.2A~~

~~Is there any Class 1, moderate energy piping in the plant?~~

Page 3.6A-24

Section 3.6.2.1.7A.2

Please clarify this paragraph. List any additional criteria used.

Page 3.6A-32

Section 3.6.2.2.3A

Justify the use of an amplification factor of less than 1.1 to account for rebound.

Pages 3.6A-33 Through 3.6A-36

Section 3.6.2.2.5A

A thrust coefficient of 0.7 was used which is less than 1.26 as specified in Section 3.6.2.2.3A and required by Standard Review Plan 3.6.2. Please provide justification for this discrepancy. Also, the load and deformation of the honeycomb panel are found without reference to the honeycomb stiffness. Please clarify these calculations.

Page 3.6A-38

Section 3.6.2.3.1A.4

Please provide clarification and justification for the use of shape factors of less than unity.

Page 3.6A-39

Section 3.6.2.3.2.2A

Please clarify the statement regarding the bottoming out of some compressive absorbers. In particular, how are restrain loads determined under these conditions?

Page 3.6A-39

Section 3.6.2.3.2.2A

Is the "retaining recess" in the bumper pipe used to restrain the moving process pipe in the lateral direction? If so, provide analysis or data to support this.

Page 3.6A-40

Section 3.6.2.3.2.2A

Provide explanation of the methods used to design omnidirectional restraints and limit stops.

Page 3.6A-40

Section 3.6.2.4A

The reference should be to MEB 3-1 not MEB-1.

Tables 3.6A-1 Through 3.6A-11

Please provide a schedule for completion of the stress analyses and updating of these tables. Will cumulative usage factors be limited to less than 1.0?

Table 3.6A-5

Please define Class 4 high-energy piping as mentioned in the footnote.

Tables 3.6A-12 Through 3.6A-20, 3.6A-27 Through 3.6A-42 and 3.6A-45 Through 3.6A-51

Provide a schedule for completion of these tables.

Figures 3.6A-12 Through 3.6A-19 and 3.6A-21 Through 3.6A-33

Provide a schedule for completion of pipe stress analyses as they effect pipe break location selections.

Page 3.6B-1

Section 3.6.2.1.1B

Is the reference to Section 3.6.2.1.1A correct?

Page 3.6B-3

Section 3.6.2.1.2.4B

Are any considerations given to relative pipe diameters when exempting impacted pipes of equal or heavier wall thicknesses from rupture? If not, provide justification for not postulating the rupture of a pipe when impacted by a pipe of larger diameter, as required by Standard Review Plan 3.6.2.

Pages 3.6B-4 and 3.6B-6

Section 3.6.2.1.2.5B

Provide justification for not postulating breaks when the cumulative usage factor is greater than 0.1 and the stress range calculated from Equation (10) is less than 3 Sm.

Page 3.6B-9

Section 3.6.2.2.1.2B

Please provide a list of all instances where crack propagation times or full break area ~~rupture~~ opening times ~~is~~ in excess of one millisecond were used.

Page 3.6B-10

Section 3.6.2.2.1.2B

Provide justification for using a thrust coefficient of less than 1.26 for saturated steam and 2.0 for subcooled water.

Page 3.6B-7

Section 3.6.2.1.2.6B, Item 5

Please provide a list of all instances where mechanistic approaches were used to reduce break areas.

~~Page 3.6B-[illegible]~~

~~Section 3.6.2.3.2.2B, Item [illegible]~~

~~[illegible] operability of components~~

~~under faulted conditions?~~

Tables 3.6B-1 Through 3.6B-4, Figure 3.6B-4

Provide a schedule for completion of needed information.

Section [illegible] through [illegible]

Please provide a schedule for completion of the failure mode analysis for pipe breaks and cracks for these systems.

Figure 3.8-4

Please provide information on the locations of all shop and field welds to process pipe in the guard pipe region. Also provide more information on the mid-guard restraint.

Section 3.9.1

Appendix 3A and Section 3.9.1.2.2B

NUREG-0800 requires that computer programs used in analyses of seismic Category I Code and non-Code items have the following information provided to demonstrate their applicability and validity:

a. The author, source, dated version and facility.
b. A description and the extent and limitations of its application.
c. Solutions to a series of test problems which shall be demonstrated to be substantially similar to solutions obtained from any one of sources 1 through 4, and source 5:

   1. Hand calculations
   2. Analytical results published in the literature
   3. Acceptable experimental tests
   4. By a MEB acceptable similar program
   5. The benchmark problems prescribed in Report NUREG/CR-1677, "Piping Benchmark Problems".

Please demonstrate compliance with these requirements and provide summary comparisons for the computer programs used in seismic Category I analyses.

Section 3.9.1.2.3B

Which computer codes were used to analyze the recirculation pumps?

Provide verification of these codes.

Section 3A.18

Please provide verification for the GHOSH-WILSON code.

Section 3A.25

Which components were used using the ANSYS code?

Table 3.9A-1

Please provide justification why turbine stop valve closures are not included in the list of transients. Also, has the alternate shutdown cooling mode been included in the design transients?

Section 3.9.1.3.1B, page 3.9B-23

Descriptions of the support and whip restraint tests could not be found. Please provide clarification as to their location.

Section 3.9.2

The discussion of the preoperational testing program does not discuss the acceptance limits for steady state and transient vibrations. What criteria will be used in developing these limits. If a stress limit will be used, what basis will be used to determine the actual stress from the measured values? Please provide a list of flow transients and a list of selected locations for visual inspections and measuring devices.

It is the staff's position that all essential safety-related instrumentaiton 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).

Section 3.9.1.1B

Standard Review Plan 3.9.2 of NUREG-0800 requires that five OBE's be assumed. The number of cycles per earthquake should be obtained from a synthetic time history (with a minimum duration of 10 seconds) used for a system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed. Please provide justification for using only 10 stress cycles or commit to using 50 maximum stress cycles.

In addition, the number of cycles for main steam and recirculation piping are missing awaiting completion of the new loads program. Please provide a schedule for completion of this information.

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 installation but not more than six months prior to initial system pre-operational testing, and should as a minimum 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 specifications.
3. Snubbers are not seized, frozen or jammed.
4. Adequate swing clearance is provided to allow snubber movement.
5. If applicable, fluid is to be 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.

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

c. Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepancies 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.

Section 3.9.2B

Please provide a statement as to the compliance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking."

Section 3.9.3

Table 3.9B-2

Much of the data included in this table will not be avaliable until the New Loads program is completed. Our review can not be completed until this information is provided. Please provide a schedule for submitting this information. Not all of the criteria for limits are included. Please provide assurance that all limits are in compliance with NUREG-0800. Also provide more detailed information on the analysis of the recirculation pump case summarized in Table 3.9B-2i.

Table 3.9B-4

NUREG-0800, Section 3.9.3, supersedes Regulatory Guides 1.67 and 1.48. Please update this table to show compliance with NUREG-0800.

Tables 3.9B-7, 8, and 9

Verify that the limits listed in these tables are in compliance with NUREG-0800, Section 3.9.3. Provide a list of all instances where the asterisked equations were used and provid the needed justification for their use.

Section 3.9.3.4A

Please provide your interpretation of jurisdictional boundaries as they pertain to NF supports. Justify your position.

Page 3.9A-26, Item 6

Please provide justification for combining vibratory loads for anchors from two component systems by the SRSS method.

Section 3.9.1.4A and Section 3.9.1.4B

This section does not address the criterion used to assure the functional capability of essential systems when they are subjected to loads in excess of those for which Service Limit B limits are specified. By essential systems are meant those ASME Class 1, 2 and 3 and any other piping systems which are necessary to shut down the plant following, or to mitigate the consequences of an accident. Please provide such criteria. In particular, have the criteria in NEDO-21985 been met?

Section 3.9.6

Section 3.9.6.2A

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 an inter-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 correction 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 to be 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, maintence, 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.