PSNH PUBLIC SERVICE
Company of New Hampshire

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October 6, 1982

SBN- 339
T.F. B 7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. Janis B. Kerrigan, Acting Chief
Licensing Branch 3
Division of Licensing

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated March 1, 1982, "Request for Additional Information" F. J. Miraglia to W. C. Tallman
(c) USNRC Letter, dated March 5, 1982, "Request for Additional Information" F. J. Miraglia to W. C. Tallman
(d) PSNH Letter, dated April 21, 1982, "Response to 210 Series RAIs; (Mechanical Engineering Branch)" J. DeVincentis to F. J. Miraglia
(e) PSNH Letter, dated April 30, 1982, "Response to 210 Series RAIs; (Mechanical Engineering Branch)" J. DeVincentis to F. J. Miraglia
(f) PSNH Letter, dated May 10, 1982, "Response to RAI 210.56; (Mechanical Engineering Branch)" J. DeVincentis to F. J. Miraglia

Subject: Revised Responses to 210 Series RAIs; (Mechanical Engineering Branch)

Dear Ms. Kerrigan:

The referenced PSNH Letters [References (d)-(f)] provided responses or commitments to respond to the 210 Series Requests for Additional Information (RAIs) which were forwarded in References (b) and (c), specifically RAI 210.3 through RAI 210.69.

B001

Meetings were conducted with the NRC Mechanical Engineering Branch on May 11-13, 1982, at the offices of United Engineers and Constructors in Philadelphia, PA during which the above RAI responses were discussed. Based on this meeting, many of the original responses have been revised for clarification, additional analysis or commitments, etc.

We have enclosed responses to RAIs 210.3 through 210.69. Original responses which have been revised have been marked "Revised", and include bars in the right margin to indicate the location of the changed information. Responses which were presented for the first time at the May 11-13, 1982 meeting have been marked "New Response at 5/11/82 Mtg".

The revised FSAR Sections which correspond to the enclosed RAI responses will be incorporated into Amendment 47.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John DeVincentis

John DeVincentis
Project Manager

ALL/dsm

Enclosures

RAI 210.3 (3.2.1, Table 3.2-2, Sheet 4)

Justify the non-seismic classification of the containment recirculating filter system. Show that its failure will not impair either the fans or ductwork.

RESPONSE:

The air cleaning or filter unit is not safety-related and is not listed as Seismic Category I. The unit is seismically anchored to restrain movement and is structurally identical to other ESF air cleaning units (containment enclosure and fuel storage building). Therefore, the unit casing will not fail during a seismic event.

Internal components of the air cleaning unit may fail structurally, but such a failure will be contained within the unit and will not impair operation of the safety-related fans, dampers and ductwork.

Revised

RAI 210.4 (3.2.1, Table 3.2-2, Sheet 1)

Explain note 9 as it applies to the reactor coolant pump flywheel.

RESPONSE:

Note 9 does not apply directly to the RCP flywheel. The flywheel is designed to maintain structural integrity during an SSE and under overspeed conditions that could result from a LOCA. A detailed discussion of the RCP flywheel is contained in FSAR Chapter 5. Sheet 1 of FSAR Table 3.2-2 ~~is~~ was ~~being~~ revised in Amendment 45 to delete the reference to Note 9 for the RCP flywheel.

RAI 210.5 (3.2.1, Page 3.2-1)

Describe methods used to confirm the structural integrity of non-seismic Category I components whose failure or collapse could result in loss of function of seismic Category I equipment.

RESPONSE:

BOP

1. For non-seismic Category I components, except piping, attachments to structural members (i.e., anchoring devices) are analyzed to demonstrate their ability to withstand applicable seismic loading. Each component's fundamental frequency in each of an X, Y and Z direction is determined, and a conservative 1.5 times the corresponding accelerations from the applicable Amplified Response Spectra (ARS) curves are the applied seismic loadings. Equivalent static analysis methods (Subsection 3.7.3.1) are then used to determine anchorage loadings and stresses. Loadings from each ear hquake direction are applied individually, and the square-root-of-the-sum-of-the-squares (SRSS) method is used for the determination of final results.

2. For those non-seismic Category I components, including piping, which are not seismically supported, the failure modes effects analysis (FMEA) performed assures that they are isolated by their location to prevent any potential impact on seismic Category I components, should there be any failure or collapse of the non-seismic Category I components or piping.

3. Also see response to RAI 220.19.

4. Isometric drawings of pipe breaks and associated pipe whip restraints for high energy lines inside containment will be provided in FSAR Amendment ~~46~~ 47.

Revised

NSSS

Consistent with Regulatory Guide 1.29, Westinghouse seismically qualifies any component whose failure could adversely effect a safety system. A specific example of such a component is the fuel handling machine located in the spent fuel pit. When seismic qualification of such components is required, they are qualified in accordance with the methods described in Sections 3.7(N) and 3.10(N) of the Seabrook FSAR. In summary, any component in Westinghouse scope which has been identified by Westinghouse as having a potential adverse impact during a seismic event on other safety-related equipment is seismically qualified.

New Response at 5/11/82 mtg

RAI 210.6 (3.2.1, Table 3.2-2, Sheets 20 and 21)

Why are the computer room system components and the primary auxiliary building dampers and ductwork not considered seismic Category I?

RESPONSE

The computer room air conditioning components are not required for safe plant shutdown. In the event of a failure of the equipment, provisions have been made to directly connect the computer room supply duct to the control room air conditioning supply air system. The latter system has sufficient capacity to furnish cooling to the computer room. The computer room ductwork is seismically supported, non-safety-related. Details of this system are contained in Section 9.4.1 and on Figure 9.4-1.

With the exception of that equipment associated with the PCCW pump area, the primary auxiliary building ventilation system is not safety-related. Where this system extends over or near safety-related equipment, the ductwork and components are seismically supported. Refer to Section 9.4.3 for further information and details.

RAI 210.7 (3.2.2, Table 3.2-2)

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:

A. Diesel Generator Fuel Oil Storage and Transfer System

   1. Remaining On-Engine Equipment and Piping

B. Diesel Generator Cooling Water System

   1. Auxiliary Coolant Pump

   2. Remaining On-Engine Equipment and Piping

C. Diesel Generator Starting System

   1. Air Compressor

   2. Remaining On-Engine Equipment and Piping

D. Diesel Generator Lubrication System

   1. Auxiliary Lube Oil Pump

   2. Remaining On-Engine Equipment and Piping

E. Diesel Generator Combustion Air Intake and Exh ust System

   1. Piping

   2. Air Intake Filter

   3. Exhaust Silencer

RESPONSE:

All of the piping and equipment associated with the diesel engine is designed to Seismic Category I requirements and is consistent with standards of Quality Group C or D of Regulatory Guide 1.26. The quality standards used for specific components are considered in compliance with Regulatory Guide 1.26, which states that systems such as diesel engine and auxiliary support systems should be designed to standards commensurate with the safety function to be performed. For the specific components identified, the following comments are noted:

A. On-engine equipment and piping is considered integral with the engine, and is designed to manufactuer's standards, which is consistent with the engine itself.

B. 1. The engine-driven pumps (coolant and lube oil) are the primary (and only) pumps required for emergency starting and operation of the diesel generator. A failure of these pumps is considered an engine failure. The auxiliary off-skid pumps are not excpected to function under emergency conditions, but could be used administratively for back-up or maintenance purposes.

2. See Response A above.

C. 1. The air compressors function is to maintain air reciever pressure between starts, but is not required for emergency starting and operation of the diesel generator. The air receiver pressure is monitored to provide ample warning for corrective actions or air compressor problems.

2. See Response A above.

D. 1. See Response B.1 above.

2. See Response A above.

E. There are no moving or rotating parts associated with the air intake filters, exhaust sinlencers, or interconnecting piping. All of these items are seismically supported within the diesel building.

Revised

RAI 210.8 (3.2.2.2, Table 3.2-2 Sheet 1)

Explain your rationale for classifying the shell side of the reactor coolant pump thermal barrier heat exchanger as ASME Code Class 3 although the tube side is Code Class 1.

RESPONSE:

Definition of the RCP thermal barrier relative to a tube side and shell side is not totally accurate. The tubes are located inside the pump casing and reactor coolant flows around the tubing. There is no shell side in the strict sense of a heat exchanger. Table 3.2-2 has been revised in Amendment 45 to correctly describe the thermal barrier and its classification.

RAI 210.9 (3.2.2.2, Table 3.2-2, Sheet 20)

Justify the absence of a quality qroup classification for the entire computer room air conditioning system.

RESPONSE:

The computer room air conditioning system is not required for safe plant shutdown, therefore, it has no ANS safety class. Refer to our answer to RAI 210.6 which is directly related to this question.

RAI 210.10 (3.2.2.2, Table 3.2-2, Sheet 20)

Control room complex emergency cleanup filter system fans and filter unit have been given ANS safety classification Non-Nuclear Safety, and the ductwork no safety classification at all. This system is considered important to safety. Provide justification for your classification.

RESPONSE:

The control room complex make-up air system from the redundant intakes (remotely spaced on opposite sides of the two units) to the control room complex emergency cleanup filter unit is ANS Safety Class 3. The placement of these air intakes essentially eliminates the possibility of having both inlets simultaneously exposed to accidently released activity. This design considerably reduces the likelihood of the emergency filter even being needed. We consider the remote air intake design to be our primary protection against the possibility of control room contamination. The filter system is, therefore, considered only as a backup and for recirculating cleanup.

The ductwork from the air cleaning unit to the associated fans and dampers, and finally to the air conditioning suppy air duct, will be upgraded to ANS Safety Class 3, Seismic Category I; FSAR Table 3.2-2, Sheet 20, will be revised to show this change.

The emergency cleaning unit fans are redundant, with Class 1E motors. While the filter units and associated fans are considered ANS Safety Class NNS, as they "can influence safe, normal operation" as defined in FSAR Subsection 3.2.2.1, they are not deemed essential to control room habitability.

As a matter of note, the overall design of the control room ventilation system was considered by the NRC in their August 1974 Safety Evaluation Report on Seabrook Station to meet the guidelines of General Design Criterion 19 with respect to potential radiation doses to control room personnel.

Revised

RAI 210.11 (3.2.2.2, Table 3.2-2)

The following ventilation systems that serve the control room or engineered safety feature rooms have portions of their systems lacking a quality group classification. Assign an appropriate quality group classification or its equivalent or justify the nonclassification:

1. Control room complex ventilation system ductwork.

2. Fuel storage building ventilation system, ventilation fans, ductwork.

RESPONSE:

1. The control room air conditioning system ductwork located within the mechanical equipment room will be classified as ANS Safety Class 3, Seismic Category I. The remaining ductwork, has no safety classification, but is seismically supported. Local failure of this ductwork will have no adverse effect on the safety-related components, equipment, or systems located in the control room complex. Table 3.2-2, Sheet 20, ~~will be changed~~ to add a note reflecting this. Section 9.4.1 ~~will be revised~~ to agree with the statements above.

2. As explained in the answer to RAI 410.36, Table 3.2-2, Sheet 7 ~~is being revised~~ (Note 12) to state that the ductwork from the downstream side of the air cleaning units to the fan intakes and the discharge of the fans to the building boundaries is Safety Class 3. This is further clarified in Section 6.5.1.

was revised in Amendment 45

RAI 210.12 (3.2.2.2, Table 3.2-2, Sheets 29, 30)

Explain the NNS ANS safety classification of the entire liquid and solid waste systems.

RESPONSE:

The liquid and solid waste systems are classified as NNS since these systems perform no safety function and are not required for safe shutdown of the reactor. Table 3.2-3 lists the quality standards applicable to these systems per Regulatory Guide 1.143. These quality standards correspond to the NNS classification.

RAI 210.13 (3.6(B).2.1, Page 3.6(B)-6)

Confirm that the "elastically calculated basis" for loadings of operating plant conditions plus an operating basis earthquake is the maximum stress as calculated by equation 9 in Paragraph NB-3652 of the ASME Code, Section III.

RESPONSE:

This criterion refers to Class 1 piping in the Seabrook plant design. Class 1 piping is all located inside containment, and therefore, the requirements of Regulatory Guide 1.46 must be met in postulating the locations of piping breaks in Class 1 piping.

The formula used in determining primary plus secondary stress intensities is equation 10 of NB-3652 of the 1971 ASME Code, Section III, with addenda up to and including Winter, 1972. This formula considers the primary plus secondary stress intensity range due to internal pressure and the range of moment loading due to thermal expansion, anchor movement, earthquake effects and other mechanical loads.

Equation 9, which is not required to be used for postulated break location, considers only the primary stress intensity due to internal pressure and moment loading due to earthquake, deadweight and other sustained design mechanical loads.

The elastically calculated basis referred to in FSAR Paragraph 3.6(B).2.1.a.1.(b) describes the criteria used to calculate the maximum stress range under the applicable load combinations, which is based upon the assumption that stresses are directly proportional to strains.

Revised

RAI 210.14 (3.6(B).2.1, Page 3.6(B)-7)

Provide drawings of all postulated pipe breaks, showing the type of break, structural barriers, restraint locations, and constrained directions in each restraint. Also provide a table showing calculated stress intensities, cumulative usage factors, and primary plus secondary stress ranges for each postulated break.

RESPONSE:

Piping drawings are not available showing structural barriers. Drawings of postulated pipe break locations and associated pipe whip restraints for high energy lines inside containment have been included in FSAR Amendment 47

Stress intensities were calculated only for Class 1 lines, using generic stress data from Westinghouse and adding UE&C-derived seismic and thermal stresses to provide very conservative values. Cumulative usage factors were assumed to exceed 0.1 at every fitting. In order to avoid performing an enormous number of calculations, we have postulated breaks at every fitting weld for Class 1 lines.

New Response at 5/11/82 mtg.

RAI 210.15 (3.6(B).2.1, Page 3.6(B)-8)

Specify where pipe whip restraints or anchors have required welding to the outer surface of the pipe. Provide details of the stress analysis performed as in the case of a riser clamp lug.

RESPONSE:

Lug attachments welded to Class 2 and 3 pipes are qualified by a procedure whose methodology is equivalent to, but more conservative than, that presented in Code Case N-318.

Local stress levels in the pipe resulting from applied lug loads are obtained by multiplying the nominal stress in the lug at the lug/pipe interface by the appropriate B or C index (as defined in Code Case N-318) for each individual loading condition. The local stresses are superimposed upon the general pipe stress as determined from program ADLPIPE to establish the total stress level in the pipe for that loading condition.

Loading conditions required to be considered for Plant Normal, Plant Upset, Plant Emergency, and Plant Faulted Operating Condition are defined (per appropriate FSAR section), and total stress in the pipe is obtained from summing the stresses for each individual loading condition that must be considered.

Local stress levels determined using B indices are added to the general stress levels from ADLPIPE and this sum is compared against allowable limits to demonstrate structural integrity. For the pipe wall, local stress levels determined using C indices are added to the general stress levels from ADLPIPE, and this sum is compared against the allowable range of stress (S_h+S_a).

Finally, weld stress is evaluated considering the absolute sum from all loads, independent of the operating condition, and compared against allowable stress from Table NF329.1-1, Subsection NF, ASME III.

Revised

RAI 210.16 (3.6(B).2.1, Page 3.6(B)-8)

Inservice inspection of break-exclusion piping must include 100% volumetric examination of all pipe welds. Augment your inservice inspection description to include this requirement at intervals shown in IWA-2400, ASME Code, Section XI.

RESPONSE:

All high energy pipe penetrations are evaluated for postulated pipe breaks. Breaks are not postulated at the main steam and feedwater penetrations. Augmented inservice inspection, including 100% volumetric examination of these penetrations, will be performed as defined in FSAR Section 6.6.

Revised

RAI 210.17 (3.6(B).2.3, Page 3.6(B)-14)

Justify the 90% of yield stress criteria in plastic restraint design. Provide examples of analysis of such a design.

RESPONSE:

Except for those components (i.e., ~~crash~~ crush pads, U-bolts) of the pipe rupture restraint (PRR) which are identified as elasto-plastic elements, the other components of the PRR structure are designed to remain elastic. The stress limit set for design of the elastic components is taken as 90% of the minimum yield strength of the material.

RAI 210.18 (3.6(B).2.3, Page 3.6(B)-15)

Provide a reference or further justification for the use of a maximum fiber strain of 50% of ultimate strain as an adequancy requirement for the load carrying capacity of piping.

RESPONSE:

We refer to ANSI/ANS 58.2 (draft) November, 1978 Section 6.3.2.a.1 for the use of this value as defining an upper-bound design limit.

Revised

RAI 210.19 (3.6(B).2.3, Page 3.6(B)-17)

What strain-rate and strain-hardening effects you have included in plastic system analysis?

RESPONSE:

The strain-hardening effect of the material has been considered in the design of U-bolt type pipe rupture restraints by utilizing an elastic-plastic bilinear stress strain curve which is approximated from either a test data stress strain curve or the minimum code yield and ultimate strength values. In the case of crush pad type pipe rupture restraints, the load-deformation characteristic of the pad is defined by the manufacturer's testing data which is basically an elastic-perfect plastic curve.

At present, the specific effect of strain-rate has not been considered. It is our criteria to use a ten percent increase in yield and ultimate strength values to account for this effect.

RAI 210.20 (3.6(B).2.5, Page 3.6(B)-18)

In order to complete our review, we must examine Appendix 3B, "Line Designation Tabulation". Provide a copy of this appendix.

RESPONSE:

Appendix 3B was incorporated into the FSAR as part of Amendment 44.

Revised

RAI 210.21 (3.6(N).2.1, Page 3.6(N)-1)

In the primary loop, what size breaks are postulated for the design of pipe whip restraints? What size breaks are postulated in the primary loop for determination of compartment pressurization and asymmetric loads? If breaks for either case are less than size, provide justification.

RESPONSE:

For the circumferential breaks postulated in the RCS, all break locations were assumed to have full double-ended breaks with the exception of the breaks postulated at the reactor vessel inlet and outlet nozzles. At these locations, the break opening area was assumed to be 144 square inches. The limited break opening area is based on the physical constraint provided by the pipe whip restraints located at the reactor vessel nozzles. Actual break opening areas based on the restraint design at these locations were calculated to be 80 square inches at the reactor vessel inlet nozzle and 30 square inches at the reactor vessel outlet nozzle. The calculated break opening areas are well below the 144 square inch break opening area assumed in the analyses.

As part of the normal design interface, UE&C has provided Westinghouse with stiffness values for the RCS pipe whip restraints. These stiffness values were then incorporated in the RCS structural evaluation to determine appropriate loading conditions on the restraints. This loading information was then transmitted to UE&C to verify the adequacy of their restraint design.

Revised

RAI 210.22 (3.6(N).2.3, Page 3.6(N)-7)

Provide a copy of test results of pipe-to-pipe impact. Also provide test results that show whipping or bending of a stainless steel pipe does not cause the section to become a missile.

RESPONSE:

Westinghouse has performed pipe whip tests demonstrating that a pipe will not break pipes of equal or greater size of the same material. These tests are documented in WCAP 7503, Supplement 1, and were submitted to the staff on the Trojan docket in response to a similar question. These test results provide justification for the Westinghouse position on this subject. It should be noted that the Westinghouse position is consistent with the staff positions identified in Regulatory Guide 1.46 and Standard Review Plan 3.6.

Revised

RAI 210.23 (3.6(N).2.5, Page 3.6(N)-11)

Review of this section shows that you have used a cumulative usage factor of 0.2 for postulated pipe rupture criteria. Branch Technical Position MEB 3-1 specifies a cumulative usage factor of less than 0.1. Provide a commitment to meet this criteria.

RESPONSE:

The number of break locations considered in the reactor coolant system using a cumulative usage factor of 0.2 is adequate. Additionally, the NRC has determined that the 0.2 criteria in WCAP-8082 (Reference 1 to FSAR Section 3.6(N) which is applicable to the Seabrook plant is acceptable. It should be noted that this item was discussed with the MEB during previous review meetings on other dockets. ~~and it was agreed that the FSAR would be revised to delete the 0.2 cumulative usage factor and replace it with a reference to WCAP-8082.~~

The FSAR has been revised in Amendment 47 to delete the 0.2 cumulative usage factor (see FSAR Section 3.6(N).2.5.a) and reference WCAP-8082.

Revised

RAI 210.24 (3.6(N).2.5, Figure 3.6(N)-2)

In addition to showing postulated break locations, they must be identified as either circumferential or longitudinal. Structural barriers, if any, restrain location and constrained directions must also be included in order to complete our review.

RESPONSE:

Break opening areas, are discussed in detail in the response to RAI 210.21. Additionally, the location of pipe whip restraints in the RCS was also discussed. ~~Westinghouse agreed to review Figure 3.6(N)-2~~ to clearly indicate the location of pipe whip restraints in the RCS. Additionally, the revised figure ~~will~~ indicates the correct location for break No. 8 in the RCS.

A new figure 3.6(N)-2 has been incorporated in FSAR Amendment 47

210.24

ATTACHMENT TO ITEM 24

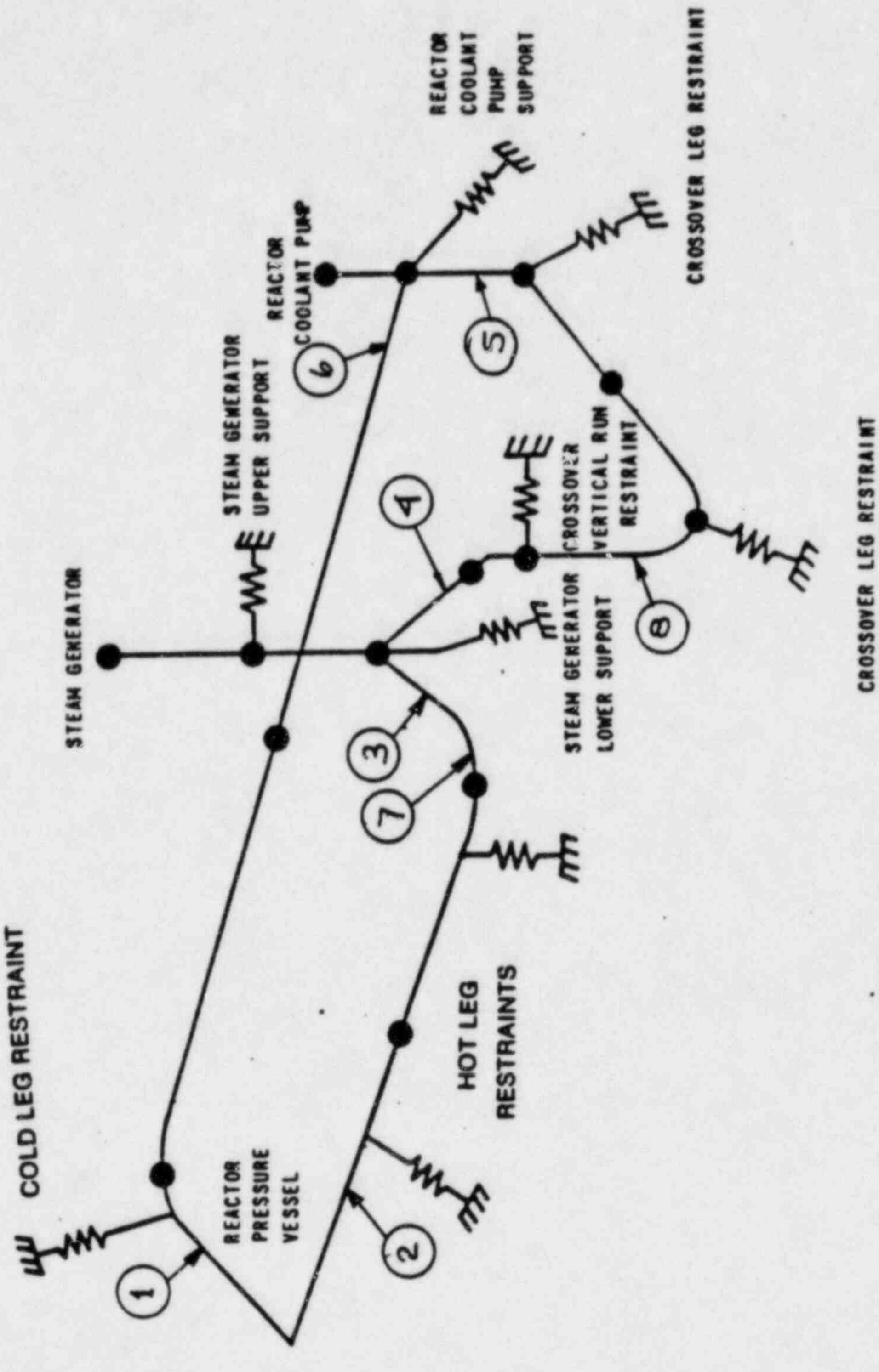

Figure 3.6(N)-2

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE SEABROOK STATION - UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	REACTOR COOLANT ~~LOOP SUPPORT~~ SYSTEM. ~~DYNAMIC STRUCTURAL MODEL~~ PIPE BREAK AND WHIP RESTRAINT LOCATIONS

RAI 210.25 (3.7.3.1, Page 3.7(B)-11)

What criteria is used to determine the number of degrees of freedom in your dynamic analysis?

RESPONSE:

BOP

Each lumped mass will have specified for it those degrees of freedom which represent the possible and/or predominant directions of motions. In some circumstances, individual masses need to be lumped for short, stiff members which exhibit rigid range behavior. An example model of a typical cable tray assembly is shown in Figure 3.7(B)-32.

NSSS

For flexible equipment, Westinghouse utilizes many degrees of freedom (e.g., 200 for steam generators) in their dynamic analysis models. It should be noted that Westinghouse assures that a sufficient number of modes is considered in the analysis, consistent with SRP 3.7.2. Westinghouse has also provided test results at a previous MEB review meeting which support their modeling techniques (e.g., for the Reactor Coolant System, NRC Docket 50-206, April 24, 1977, "San Onofre Nuclear Generating Station Seismic Re-evaluation and Modifications"), and additional data on tanks, valves, and typical piping systems.

For auxiliary mechanical equipment with natural frequencies below 33 Hz, test results were presented at a previous MEB review meeting to support the number of modes considered in the analyses.

The above information, which was presented to the NRC at previous MEB review meetings, is applicable to the Seabrook plant.

New Response at 5/11/82 mtg.

RAI 210.26 (3.7.3.1, Page 3.7(B)-11)

Demonstrate that the equivalent static load method analysis you have used accounts for relative motion between all parts of support.

RESPONSE:

BOP

When significant relative motions among the parts of any supporting system are encountered, their effects are determined statically and superimposed with other analytical results associated with any particular dynamic event.

NSSS

The equivalent static load method has not been used on any Westinghouse piping and piping supports.

New Response at s/l/k mtg

RAI 210.27 (3.9(B).1.1, Page 3.9(B)-1)

Are there any reactor coolant pressure boundary, ASME Code Class 1 or CS components in BOP? If so, provide or reference an appropriate design transient list.

RESPONSE:

BOP

Yes; see Subsection 3.9(N).1.1 for design transient list applicable to BOP Class 1 components.

Class 1 lines in the BOP scope are identified below:

Line No.	Line Size	P&ID
91-1	1"	9763-F-805002
91-2	1"	9763-F-805002
328-6	2"	9763-F-805003
328-7	1½"	9763-F-805003
328-10	3/4"	9763-F-805003
329-4	2"	9763-F-805004
329-5	1½"	9763-F-805004
329-8	3/4"	9763-F-805004
330-4	2"	9763-F-805005
330-5	1½"	9763-F-805005
330-6	3/4"	9763-F-805005
331-4	2"	9763-F-805006
331-5	1½"	9763-F-805006
331-8	3/4"	9763-F-805006
80-1	6"	9763-F-805007
80-2	3"	9763-F-805007
80-6	3"	9763-F-805007
74-1	6"	9763-F-805007
75-1	6"	9763-F-805007
76-1	6"	9763-F-805007

Revised

NSSS

Westinghouse has responsibility for Class 1 component core support structures and specific Class 1 piping. UE&C has responsibility for pressurizer safety relief line, the reactor coolant system drain line and Class 1 reactor coolant pump seal piping.

New Respon 2t 5/11/8? mtg

RAI 210.28 (3.9(B).1.2, Page 3.9(B)-1)

NUREG-0800 requires that computer programs 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 limitation 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 an MEB acceptable similar program.

   5. The benchmark problems prescribed in Report NUREG/CR-1677, "Piping Benchmark Problems".

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

RESPONSE:

BOP

The above information is documented and available for review for all structural analysis computer programs used by UE&C. The verification package for the in-house version of ADLPIPE has been supplemented by Problem #4 from NUREG/CR-1677. Summary comparisons show excellent agreement. (A copy of the supplemental verification is provided by a separate transmittal.*) Revisions to FSAR Subsection 3.9(B).1.2 which describes the verification methods used for each structural computer program have been provided in Amendment 45.

Revised

NSSS

The computer codes used by W for Class 1 analyses are described in FSAR Subsection 3.9(N).1.2 and in References 1 (WCAP-8252) and 2 (WCAP-8929) to Section 3.9(N). WCAP-8252 has been approved by the NRC, and WCAP-8929 is currently undergoing review. This information has been sufficient to address this concern during previous MEB review meetings. ~~The NRC considers this information satisfactory at this time.~~

New Response at [illegible]

* PSNH Letter, dated August 17, 1982, "ADLPIPE Benchmark; RAI 210.28; (Mechanical Engineering Branch)" J. DeVincentis to F. J. Miraglia

Revised

RAI 210.29 (3.9(B).1.4, Page 3.9(B)-6)

Where is AISC criteria used in evaluation of faulted conditions? Justify its use.

RESPONSE:

The AISC criteria were used in the support designs of the following mechanical components:

| *

Containment Spray Pumps

Primary Component Cooling Water Pumps

Spent Fuel Pool Pumps

Primary Component Cooling Water Head Tank

Cation Bed Demineralizer Tank

Mixed Bed Demineralizer Tank

Containment Spray Heat Exchanger

Spent Fuel Pool Cooling Heat Exchanger

Emergency Feed Pumps

(See the response to RAI 210.39 for justification).

* Service Water Strainers were deleted

RAI 210.30 (3.9(B).1.4, Page 3.9(B)-7) *

This section does not address the criteria 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. Provide such criteria.

RESPONSE: *

~~Design service limits, as defined in NCA2142, are not identified in the design specification for piping.~~

~~For Seabrook, design conditions are specified to be normal, upset, emergency and faulted, as defined in the ASME Code, 1971 edition, with addenda up to and including Winter, 1972. The criteria used to assure the functional capability of essential systems to shut down the plant safely following, and to mitigate the consequences of an accident, are described in FSAR Subsections 3.9(B).1.4.b.1 (a), (b) and (c).~~

*Deleted by NRC at 5/11/82 meeting

Revised

RAI 210.31 (3.9(B).2.1, Page 3.9(B)-8)

What are the acceptance limits for steady state and transient vibration? The program must include a list of different flow modes and a list of selected locations for visual inspection and measurements.

RESPONSE:

The preoperational and startup vibration test program is under development and is scheduled to be available in October, 1982.

Also, see revised FSAR paragraphs 3.9(B).2.1.a.1 and 3.9(B).2.1.a.5 (Amendment ~~46~~ 47).

Revised

RAI 210.32 (3.9(B).2.1, Page 3.9(B)-10)

What Code-allowable stress limits are used for acceptability of motion due to dynamic effects?

RESPONSE:

See revised FSAR paragraph 3.9(B).2.1.c (Amendment ~~46~~ 47).

Revised

RAI 210.33 (3.9(B).3.1, Page 3.9(B)-13)

What piping systems are not designed according to ASME Section III? What design criteria was used for these systems?

RESPONSE:

Any non-safety related system is designated NNS. All piping systems designated as NNS are designed to ANSI-B31.1 requirements. Item 4 of ~~Page~~ 3.9(B).3.1.b.4 is being revised in Amendment ~~43~~ 47 to delete "Category I".

FSAR Subsection

Revised

RAI 210.34 (3.9(B).3.3, Page 3.9(B)-22)

Regulatory Guide 1.67 does not address closed systems or systems with a water slug. How was 1.67 used for the installation and design of pressure relief devices?

RESPONSE:

Although Regulatory Guide 1.67 does not address closed systems, Subsection 3.9(B).3.3 of the FSAR addresses the evaluation of safety and relief valves for closed and open systems.

The only relief valves presently having a water seal which could introduce a water slug are the pressurizer relief valves. However, an alternate piping layout has been developed and will be implemented to eliminate the water slug concern. Paragraph 3.9(B).3.3a ~~will be~~ was revised in Amendment 45 as follows: ~~to reflect this change:~~

"The three safety valves are mounted on the pressurizer nozzles with the short inlet pipe and elbow necessary to position the valves vertically. The total length of pipe, elbow and weld neck flange is approximately 24 inches and is as short as possible to minimize the pressure drop on the inlet side of the valve.

The two power operated relief valves have inlet piping shaped to form a water seal below each valve seat to reduce the problem of steam and hydrogen leakage through the valve seats. When the valves open, water from the seals is discharged ahead of the steam as the valve disc lifts. The dynamic effects from the flow of water and steam are included in the design analysis."

RAI 210.35 (3.9(B).3.3, Page 3.9(B)-23)

Was Regulatory Guide 1.67 used to determine the spacing of the safety valves on the main steam lines?

RESPONSE:

Spacing of the safety valves on the main steam lines is in compliance with Regulatory Guide 1.67 and referenced Code Case 1569.

Revised

RAI 210.36 (3.9(B).3.3, Page 3.9(B)-24)

Provide a schedule for completion of dynamic analyses results.

RESPONSE:

The information now indicated as "later" on FSAR Tables 3.9(B)-19 and 3.9(B)-20 will be provided ~~by September 15, 1982.~~ in a future amendment.

RAI 210.37 (3.9(B).3.4), Page 3.9(B)-26)

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

RESPONSE:

BOP

Jurisdictional boundaries of supports designed and fabricated to Subsection NF requirements as shown in NF-1000 for plates, welding and bolting is as follows:

| Revised

1. Plates

   A. Support plates that are embedded in concrete with integral embedded anchors (studs) do not fall under NF jurisdication, whether or not they protrude from the surface of concrete.

   B. Loose or adjustable base plates which only support compressive loads do not fall under NF jurisdiction.

   C. Loose plates that are welded to component supports do fall under NF jurisdiction. (Surface mounted plates)

2. Welding

   A. The weld used to attach NF supports to building steel, supplementary steel or intervening members is considered to fall within the jurisdiction of NF.

3. Bolting

   A. Embedded custom-designed anchor bolts are designed and purchased to AISC requirements and the additional materials, Certification and NDE Examination Requirements of ASME Subsection NF.

   B. Standard expansion anchors which are manufactured and stocked as catalogue items such as hilti-kwik bolts, fall under the jurisdication of Subsection NF.

NSSS

The Class 1 component supports supplied by Westinghouse for Seabrook are consistent with the requirements of Subsection NF of the ASME Code. Westinghouse supplies Class 1 supports from the base plate or concrete to the component. Therefore, the jurisdictional boundary for Westinghouse supplied Class 1 supports is well defined.

Class 2 and 3 jurisdictional boundaries were also discussed. Design criteria for Class 2 and 3 component supports are described in FSAR Subsection 3.9(B).3.4.

New Respo at 5/11/8 mtg

Class 2 and 3 component supports are generally attached to the component and the base plate.

The BOP designer is responsible for anchoring the component to the supporting structure. Therefore, as in the case of Class 1, the jurisdictional boundary for Class 2 and 3 supports is well defined.

Revised

RAI 210.38 (3.9(B).3.4, Page 3.9(B)-26)

Provide an example of the analysis performed on ASME Code Class 1, 2 and 3 valve supports.

RESPONSE:

The only valve supports that were analyzed as valve supports were the pressurizer safety and relief valve supports. These are ASME Class 1 supports, and the stress report ~~has not yet been completed.~~ will be provided in a future amendment.

RAI 210.39 (3.9(B).3.4, Page 3.9(B)-26)

The design criteria used for mechanical equipment supports needs clarification. Subsection NF, ASME Code, Section III is applicable to these supports. Justify the use of AISC allowable stresses to demonstrate that your design criteria satisfy the requirements of Subsection NF.

RESPONSE:

BOP

The supports of certain mechanical equipment purchased circa 1974 were designed in accordance with the requirements defined in the AISC Manual of Steel Construction. In addition, the following criteria were included in the support designs:

1) Material properties used in conjunction with the support design were obtained from the tables for material strength values in the ASME III, Subsection NA, Appendix I.

2) The allowable bolt stresses were derived from the AISC Specification, without use of one-third increase factor for Normal and Upset Conditions. For the faulted condition the AISC allowable of 0.6 Fy was multiplied by the strength factors noted in SRPs 3.8.3 and 3.8.4.

3) The loading considered in the design of the supports and anchor bolts are the same as those imposed on the components. More specifically, the appropriate loads are applied to the components and the resulting reactions are used to design the supports.

4) For the faulted condition, tensile and bending stresses were limited to 90% of the material yield strength and shear stresses were limited to 60% of the material yield strength which compare favorably with the limits defined by ASME III, Subsection NF, for faulted conditions.

5) Buckling evaluations were performed in accordance with the AISC criteria without use of increase factor for faulted conditions.

6) The highest value of KL/R is less than 20 for all mechanical components (excluding piping systems).

The following tabulation shows the stress limits used for various bolt materials:

1) Stress Limits for Anchor Bolts for Equipment

Bolt Material	Allowable Tensile Stress	Allowable Shear Stress
ASTM A193 Grade B7		
Under 2-1/2" Ø Fy = 105 ksi Fu = 125 ksi	Ft = 0.6 Fy 0.5 Fu = 62.5 ksi	Fv = 0.4 Fy = 42 ksi

Revised

(7.10.3)

ASTM A540 Grade B23 Class 4

Up to 3"Ø		
Fy = 120 ksi	Ft = 0.6 Fy 0.5 Fu	Fv = 0.4 Fy
Fu = 135 ksi	= 67.5 ksi	= 48 ksi

ASTM A354 Grade BD

For 1/4" to 2-1/2"Ø		
Fy = 130 ksi	Ft = 0.6 Fy 0.5 Fu	Fv = 0.4 Fy
Fu = 150 ksi	= 75 ksi	= 52 ksi

2) High strength bolts for equipment on structural steel and for steel-to-steel connections

ASTM A325

1/2" to 1"Ø	1-1/8" to 1-1/2"Ø
Fy = 92 ksi	Fy = 81 ksi
Fu = 120 ksi	Fu = 105 ksi

ASTM A490

1/2" to 1-1/2"Ø
Fy = 130 ksi
Fu = 150 ksi

All allowable tension and shear values are in accordance with Manual of Steel Construction - AISC.

For the faulted condition, the strength factors of 1.6 or 1.7 as noted in SRP 3.8.3 and 3.8.4 were applied to the above.

NSSS

The Class 1 component supports supplied by Westinghouse for Seabrook are consistent with the requirements of Subsection NF of the ASME code. Westinghouse supplies Class 1 supports from the base plate or concrete to the component. Therefore, the jurisdictional boundary for Westinghouse supplied Class 1 supports is well defined.

Class 2 and 3 jurisdictional boundaries were also discussed. Design criteria for Class 2 and 3 component supports are described in FSAR Subsection 3.9(N).3.4. Class 2 and 3 component supports are generally attached to the component and the base plate.

The BOP designer is responsible for anchoring the component to the supporting structure. Therefore, as in the case of Class 1, the jurisdictional boundary for Class 2 and 3 supports is well defined.

New Response at 5/11/82 mtg.

RAI 210.40 (3.9(B).3.4, Page 3.9(B)-26)

Provide design criteria for any snubbers.

RESPONSE:

BOP

A revised FSAR Subsection 3.9(B).3.4 which incorporates snubber design criteria has been provided in FSAR Amendment 45. Revised

NSSS

The only snubbers supplied by Westinghouse are located at the steam generator upper support points. These snubbers are analyzed in accordance with the criteria described in Section 3.9(N).1 for Class 1 component supports. Additional information for these snubbers is provided in FSAR Section 5.4.14. New Response at 5/11/82 mtg

Attachment A

NRC STAFF COMMENTS ON INSERVICE PUMP AND VALVE TESTING PROGRAMS AND RELIEF REQUESTS

The NRC staff, after reviewing a number of pump and valve testing programs, has determined that further guidance might be helpful to illustrate the type and extent of information we feel is necessary to expedite the review of these programs. We feel that the Licensee can, by incorporating these guidelines into each program submittal, reduce considerably the staff's review time and time spent by the Licensee in responding to NRC staff requests for additional information.

The pump testing program should include all safety related* Class 1, 2 and 3 pumps which are installed in water cooled nuclear power plants and which are provided with an emergency power source.

The valve testing program should include all the safety related valves in the following systems excluding valves used for operating convenience only, such as manual vent, drain, instrument and test valves, and valves used for maintenance only.

PWR

a. High Pressure Injection System.

b. Low Pressure Injection System.

c. Accumulator Systems.

d. Containment Spray System.

e. Primary and Secondary System Safety and Relief Valves.

f. Auxiliary Feedwater Systems.

g. Reactor Building Cooling System.

h. Active Components in Service Water and Instrument Air Systems which are required to support safety system functions.

i. Containment Isolation Valves required to change position to isolate containment.

*Safety related - necessary to safely shut down the plant and mitigate the consequences of an accident.

210.41 (3.9(B).6) As required by 10 CFR 50.55 a(g), we request that you submit your preservice and initial 120 month inservice testing program for pumps and valves. Attachment A provides a suggested format for this submittal and a discussion of information we require to justify any relief requests.

RESPONSE: The scope of pumps and valves operability testing, including adherence to ASME Boiler and Pressure Vessel Code, Section XI and 10 CFR 50.55 a(g), is provided in FSAR Section 3.9(B).6. 10 CFR 50.55 a(g) does not specify a submittal date for a preservice or inservice testing program for pumps and valves; however, the pump and valve test program will be submitted, with the Inservice Inspection Program, within six months of the anticipated date for commercial operation. This submittal date is consistent with previous NRC staff guidance.

(2) -

j. Chemical & Volume Control System.

k. Other key components in Auxiliary Systems which are required to directly support plant shutdown or safety system function.

l. Residual Heat Removal System.

m. Reactor Coolant System.

BWR

a. High Pressure Core Injection System.

b. Low Pressure Core Injection System.

c. Residual Heat Removal System (Shutdown Cooling System).

d. Emergency Condenser System (Isolation Condenser System).

e. Low Pressure Core Spray System.

f. Containment Spray System.

g. Safety, Relief and Safety/Relief Valves.

h. RCIC (Reactor Core Isolation Cooling) System.

i. Containment Cooling System.

j. Containment isolation valves required to change position to isolate containment.

k. Standby liquid control system (Boron System)

l. Automatic Depressurization System (any pilot or control valves, associate hydraulic or pneumatic systems, etc.)

m. Control Rod Drive Hydraulic System ("Scram" function)

n. Other key components in Auxiliary Systems which are required to direct support plant shutdown or safety system function.

o. Reactor Coolant System.

1210.41

Inservice Pump and Valve Testing Program

I. Information required for NRC Staff Review of the Pump and Valve Testing Program

A. Three sets of P&ID's, which include all of the systems listed above, with the code class and system boundaries clearly marked. The drawings should include all of the components present at the time of submittal and a legend of the P&ID symbols.

B. Identification of the applicable ASME Code Edition and Addenda.

C. The period for which the program is applicable.

D. Identify the component code class.

E. For pump testing: Identify

1. Each pump required to be tested (name and number)

2. The test perameters to be measured

3. The test frequency

F. For valve testing: Identify

1. Each valve in ASME Section XI Categories A & B that will be exercised every three months during normal plant operation (indicate whether partial or full stroke exercise, and for power operated valves list the limiting value for stroke time.)

2. Each valve in ASME Section XI Category A that will be leak tested during refueling outages (Indicate the leak test procedure you intend to use)

3. Each valve in ASME Section XI Categories C, D and E that will be tested, the type of test and the test frequency. For check valves, identify those that will be exercised every 3 months and those that will only be exercised during cold shutdown or refueling outages.

II. Additional Information That Will Be Helpful in Speeding Up the Review Process

A. Include the valve location coordinates or other appropriate location information which will expedite our locating the valves on the P&ID's.

B. Provide P&ID drawings that are large and clear enough to be read easily.

(2.0.4

C. Identify valves that are provided with an interlock to other components and a brief description of that function.

Relief Requests from Section XI Requirements

The largest area of concern for the NRC staff, in the review of an inservice valve and pump testing program, is in evaluating the basis for justifying relief from Section XI Requirements. It has been our experience that many requests for relief, submitted in these programs, do not provide adequate descriptive and detailed technical information. This explicit information is necessary to provide reasonable assurance that the burden imposed on the licensee in complying with the code requirements is not justified by the increased level of safety obtained.

Relief requests which are submitted with a justification such as "Impractical", "Inaccessible", or any other categorical basis, will require additional information, as illustrated in the enclosed examples, to allow our staff to make an evaluation of that relief request. The intention of this guidance is to illustrate the content and extent of information required by the NRC staff, in the request for relief, to make a proper evaluation and adequately document the basis for that relief in our safety evaluation report. The NPC staff feels that by receiving this information in the program submittal, subsequent requests for additional information and delays in completing our review can be considerably reduced or eliminated.

I. Information Required for NRC Review of Relief Requests

A. Identify component for which relief is requested:

1. Name and number as given in FSAR

2. Function

3. ASME Section III Code Class

4. For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000

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

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

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

E. Provide the schedule for implementation of the procedure(s) in (D).

(3.10.4)

II. <u>Examples to Illustrate Several Possible Areas Where Relief May Be Granted and the Extent and Content of Information Necessary to Make An Evaluation</u>

A. Accessibility: The regulation specifically grants relief from the code requirement because of insufficient access provisions. However, a detailed discussion of actual physical arrangement of the component in question to illustrate the insufficiency of space for conducting the required test is necessary.

Discuss in detail the physical arrangement of the component in question to demonstrate that there is not sufficient space to perform the code required inservice testing.

What alternative surveillance means which will provide an acceptable level of safety have you considered and why are these means not feasible?

B. Environmental Conditions (e.g., High radiation level, High temperature, High humidity, etc.)

Although it is prudent to maintain occupation radiation exposure for inspection personnel as low as practicable, the request for relief from the code requirements cannot be granted solely on the basis of high radiation levels alone. A balanced judgment between the hardships and compensating increase in the level of safety should be carefully established. If the health and safety of the public dictates the necessity of inservice testing, alternative means or even decontamination of the plant if necessary should be provided or developed.

Provide additional information regarding the radiation levels at the required test location. What alternative testing techniques which will provide an acceptable level of assurance of the integrity of the component in question have you considered and why are these techniques determined to be impractical?

C. Instrumentation is not originally provided

Provide information to justify that compliance with the code requirements would result in undue burden or hardships without a compensating increase in the level of plant safety. What alternative testing methods which will provide an acceptable level of safety have you considered and why are these methods determined to be impractical?

D. Valve Cycling During Plant Operation could put the Plant in an Unsafe Condition

The licensee should explain in detail why exercising tests during plant operation could jeopardize the plant safety.

(2), e.

E. Valve Testing at Cold Shutdown or Refueling Intervals in lieu of the 3 Month Required Interval

The liensee should explain in detail why each valve cannot be exercised during normal operation. Also, for the valves where a refueling interval is indicated explain in dedail why each valve cannot be exercised during cold shutdown intervals.

III. Acceptance Criteria for Relief Request

The Licensee must successfully demonstrate that:

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

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

Standard Format

A standard format, for the valve portion of the pump and valve testing program and relief requests, is included as an attachment to this Guidance. The NRC staff believes that this standard format will reduce the time spent by both the staff in our review and by the licensee in their preparation of the pump and valve testing program and submittals. The standard format includes examples of relief requests which are intended to illustrate the application of the standard format and are not necessarily a specific plan relief request.

ATTACHMENT

STANDARD FORMAT

VALVE INSERVICE TESTING PROGRAM SUBMITTAL

Class	Coordinates	Valve Category A	B	C	D	E	Size (inches)	Valve Type	Actuator Type	Normal Position	Test Requirements	Relief Requests*	Testing Alternative	Remarks (Not to be used for relief basis)
3	D-14					X	4	GA	M	LO	ET			
3	D-15				X		6	DE	NA	C	DT			
3	C-15			X			16	CK	SA	-	CV	X	CS	
3	C-15			X			16	CK	SA	-	CV			
3	E-14			X			3	REL	SA	-	CV			
3	D-11		X			X	4	GL	M	C	O	X	ET	
											MT			60 sec.
3	B-11			X			3/4	REL	SA	-	SRV			
3	B-11			X			3/4	REL	SA	-	SRV			
2	A-10			X			3	REL	SA	-	SRV			
2	B-10			X			3	REL	SA	-	SRV			
2	D-14	X					10	GA	MO	C	O			
											LT	X		
											MT			30 sec.

Legend for Valve Testing Example Format

Q - Exercise valve (full stroke) for operability every (3) months.

LT - Valves are leak tested per Section XI Article IWV-3420.

MT - Stroke time measurements are taken and compared to the stroke time limiting value per Section XI Article IWV 3410.

CV - Exercise check valves to the position required to fulfill their function every (3) months.

SRV - Sefety and relief valves are tested per Section XI Article IWV-3510.

DT - Test Category D valves per Section XI Article IWV-3600.

ET - Verify and record valve position before operations are performed and after operations are completed, and verify that valve is locked or sealed.

CS - Exercise valve for operability every cold shutdown.

RR - Exercise valve for operability every reactor refueling.

Relief Request Basis

System: Auxiliary Coolant System, Component Cooling

1.	Valve:	717
	Category:	C
	Class:	3
	Function:	Prevent backflow from the reactor coolant pump cooling coils.
	Impractical Test Requirement:	Exercise valve for operability every three months.
	Basis for relief:	To test this valve would require interruption of cooling water to the reactor coolant pumps motor cooling coils. This action could result in damage to the reactor coolant pumps and thus place the plant in an unsafe mode of operation.
	Alternative Testing:	This valve will be exercised for operability during cold shutdowns.
2.	Valve:	834
	Category:	B-E
	Class:	3
	Function:	Isolate the primary water from the component cooling surge tank during plant operation. It is normally in the closed position, but routine operation of this valve will occur during during refueling and cold shutdowns.
	Impractical Test Requirement:	Exercise valve (full stroke) for operability every three (3) months.
	Basis for Relief:	This valve is not required to change position during plant operation to accomplish its safety function. Exercising this valve will increase the possibility of surge tank line contamination.
	Alternate Testing:	Verify and record valve position before and after each valve operation.

3. Valve: 744B

Category: A

Class: 2

Function: Isolate the residual heat exchangers from the leg R.C.S. backflow and accumulator backflow.

Test Requirements: Seat leakage test.

Basis for Relief: This valve is located in a high radiation field (2000 mr/hr) which would make the required seat leakage test hazardous to test personnel. We intend to seat leak test two other valves (875B and 876B) which are in series with this valve and will also prevent backflow. We feel that by complying wiht the seat leakage requirements we will not achieve a compensatory increase in the level of safety.

Alternative Testing: No alternative seat leak testing is proposed.

RAI 210.42 (3.9(N).1.2, Page 3.9(N)-20)

Provide references 1 and 2 for our review.

RESPONSE:

Reference 1 (WCAP-8252) has been reviewed and approved by the NRC. Reference 2 (WCAP-8929) has been submitted to the NRC and is currently being reviewed by the NRC and Oak Ridge National Labs.

Revised

RAI 210.43 (3.9(N).1.4, Page 3.9(N)-33)

How is the critical buckling strength for component supports determined?

RESPONSE:

Westinghouse performs buckling analysis in accordance with the requirements of the ASME Code, Section III, Appendix F, and meets the 2/3 critical buckling criteria. Subsection ~~3.9(N).1.4~~ 3.9(N).1.4.9.3 of the FSAR ~~will be~~ has been revised in Amendment #7 to delete the exception to Appendix F which currently states that 90% of critical buckling will be met.

210.43

ATTACHMENT TO ITEM 43

method data at the loads resulting from the system analysis will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

DELETE

~~3. Component Support Buckling Allowable Load~~

~~In the design of component supports, members compressive axial loads are limited to 0.67 times the critical buckling strength. If, as a result of more detailed evaluation of the supports the member compressive axial loads can be shown to safely exceed 0.67 times the critical buckling strength for the faulted condition, verification of the support functional adequacy will be documented and submitted to the NRC for review. The member compressive axial loads will not exceed 0.67 times the critical buckling strength without NRC acceptance. In no case will the compressive load exceed 0.9 times the critical buckling strength.~~

Loading combinations and allowable stresses for ASME III Class 1 components and supports are given in Tables 3.9(N)-2 and 3.9(N)-3. For Faulted condition evaluations, the effects of the safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) are combined using the square-root-of-the-sum-of-the-squares (SRSS) method. Justification for this method of load combinations is contained in References (4) and (5).

3.9(N).2 Dynamic Testing and Analysis

3.9(N).2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

A preoperational piping vibrational and dynamics effects testing program will be conducted for the reactor coolant loop/supports system during preoperational testing. The purpose of these tests will be to confirm that the system has been adequately designed and supported for vibration, as required by Section III of the ASME Code, paragraph NB-3622.3. The tests will include reactor coolant pump starts and trips. If vibration is experienced, which, from visual observation, appears to be excessive, either: 1) an instrumented test program on the piping, will be conducted and the system reanalyzed to demonstrate that the observed levels will not cause ASME Code stress and fatigue limits to be exceeded, 2) the cause of the excessive vibration will be eliminated, or 3) the support system will be modified to reduce the vibration. Particular attention will be provided at those locations where the vibration is expected to be the most severe for the particular transient condition being studied.

It should be noted that the layout, size, etc., of the reactor coolant loop and surge line piping used in the Seabrook plants is very similar to that employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant

RAI 210.44 (3.9(N).2.5, Pages 3.9(N)-38 to 44)

Previous analysis 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 nozzles to piping terminal ends.

    c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

(210.4A)

Revised

4. 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.

5. 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.

6. For all analyses performed, included 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.

7. 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.

8. 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.

RESPONSE:

Westinghouse, in its analyses of reactor system components and their supports, has considered asymmetric LOCA loadings in the Seabrook plant design and analysis. The analysis methods used by Westinghouse are consistent with NUREG-0609. This question is addressed in Seabrook FSAR Subsections 3.9(N).2.5, 3.9(N).3 and the revised/updated Subsections 3.9(N).1.2, 3.9(N).1.4b, 3.9(N).1.4c, 3.9(N).1.4d, 3.9(N).1.4e, 3.9(N).1.5, 3.9(N).1.6, 3.9(N).1.7, 3.9(N).4.2, 3.9(N).4.3 and 3.9(N).4.4 ~~which will be provided later.~~ which are provided in Amendment 47.

*Blowdown jet forces at the location of 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.

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ATTACHMENT TO RESPONSE TO QUESTION 210.44

in the fatigue usage factor caused by these tests is easily covered by the conservative number (200) of primary side leakage tests that are considered for design.

2. Secondary Side Hydrostatic Test

The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F, coincident with the primary side at 0 psig.

For design purposes, it is assumed that the steam generator will experience 10 cycles of this test.

These tests may be performed either prior to plant startup, or subsequently following shutdown for major repairs or both.

3.9(N).1.2 Computer Programs Used in Analyses

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of seismic Category I components and equipment. These are described and verified in References (1) and (2).

a. WESTDYN-7

Static and dynamic analysis of redundant piping systems.

b. FIXFM

Time history response of three-dimensional structures.

c. WESDYN-2

Piping system stress analysis from time history displacement data.

d. STHRUST

Hydraulic loads on loop components from blowdown information.

e. WESAN

Reactor coolant loop equipment support structures analysis and evaluation.

f. WECAN

Finite element structural analysis.

g. DARIWOSTAS

Dynamic transient response analysis of reactor vessel and internals.

3.9(N).1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for seismic Category I systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests, as discussed in Subsection 3.9(N).2.

3.9(N).1.4 Considerations for the Evaluation of the Faulted Condition

a. Loading Conditions

The structural stress analyses performed on the reactor coolant system consider the loadings specified as shown in Table 3.9(N)-2. These loads result from thermal expansion, pressure, weight, Operating Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE), design basis loss of coolant accident, and plant operational thermal and pressure transients.

b. Analysis of the Reactor Coolant Loop and Supports

INSERT A

The loads used in the analysis of the reactor coolant ~~loop piping~~ loop/support system are described in detail below.

1. Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at changes in direction or flow area.

2. Weight

A weight analysis is performed to meet Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

3. Seismic

The forcing functions for the reactor coolant loop piping seismic analyses are three orthogonal components of

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INSERT A (Modification to 3.9(N).1.4b)

The reactor coolant loop piping is evaluated in accordance with the Criteria of ASME III, NB-3650 and Appendix F. The loads included in the evaluation result from the SSE, dead weight, pressure, and LOCA loadings (loop hydraulic forces, asymmetric sub-compartment pressurization forces, and reactor vessel motion).

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The component upper and lower lateral supports are inactive during plant heatup, cooldown and normal plant operating conditions. However, these restraints become active when the plant is at power and under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The total response is obtained directly by direct time integration of the equations of motion. The results of the time history analysis are forces and displacements. The time history displacement response is then used in computing support loads and in performing the reactor coolant loop piping stress evaluation.

3. Loss of Coolant Accident

The mathematical model used in the static analyses is modified for the loss of coolant accident analyses to represent the severance of the reactor coolant loop piping at the postulated break location. Modifications include addition of the mass characteristic of the piping and equipment. To obtain the proper dynamic solution, two masses, each containing six dynamic degrees of freedom and located one on each side of the break, are included in the mathematical model. The natural frequencies and eigenvectors are determined from this broken loop model.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full power loss of coolant accident and steam line break is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this non-linear element is treated as an externally applied force in the overall normal mode solution. Multiple non-linear elements can be applied at the same node, if necessary.

The time-history solution is performed in subprogram FIXFM3. The input to this subprogram consists of the natural frequencies, normal modes, applied forces and nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN-7 program. To properly simulate the release of the strain energy

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in the pipe, the internal forces in the syste the postulated break location due to the initial steady-stat ydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the olution is only for the transient problem (the dynamic respo e of the system from the static equilibrium position). The time history displacement solution of all dynamic degrees of freedom is obtained using subprogram FIXFM3 and employing 4 percent critical damping.

The loss of coolant accident displacements of the reactor vessel are applied in time history form as input to the dynamic analysis of the reactor coolant loop. The loss o coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions, cavity pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Subsection 3.9(N).1.4f.

INSERT) B

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

~~The support loads are computed by multiplying the support stiffness matrix and the displacement vector at the support point. The support loads are used in the evaluation of the supports.~~

The time-history displacements of the FIXFM3 subprogram are used as input to WESDYN-2 to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses. ~~The results of this solution are used in the piping stress evaluation.~~

4. Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts, a uniform, a linear, and a non-linear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the non-linear portion causes a skin stress.

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The resultant asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure build up in the loop compartments, are applied to the same integrated RCL/supports system model used to compute loadings on components, component supports and RCL piping as discussed above. The response of the entire system is obtained for the various external pressure loading cases from which the internal member forces and piping stresses are calculated for each pipe break case considered. The equipment support leads and piping stresses resulting from the external pressure loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The break locations considered for subcompartment pressurization are those postulated for the RCL LOCA analysis, as discussed in Section 3.6N and WCAP-8172 (Ref 1 of Section 3.6N). The asymmetric subcompartment pressure loads are provided to Westinghouse by United Engineers & Constructors. The analysis to determine these loads is discussed in Section 6.2.

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For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors and cumulative usage factors are calculated. The WESTDYN-7 program is used to perform this analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3650. Since it is impossible to predict the order or occurrence of the transients over a forty-year life, it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $< 10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

d. <u>Primary Component Supports Models and Methods</u>

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The equipment support structure models are dual-purpose since they are required to: 1) quantitatively represent the elastic restraints which the supports impose upon the loop, and 2) evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

~~Models for the STRUDL computer program are constructed for the steam generator lower, steam generator upper lateral, reactor coolant pump lower and pressurizer supports. The reactor vessel supports are modeled using the WECAN computer program. Structure geometry, topology and member properties are used in the modeling.~~

A description of the supports is found in Subsection 5.4.14. Detailed models are developed using beam elements and plate elements, where applicable. (INSERT C)

The respective computer programs are used with these models to obtain support stiffness matrices and member influence coefficients for the steam generator, reactor coolant pump, pressurizer and reactor vessel supports. Unit force along and unit moment about each coordinate axis are applied to the models at the equipment vertical centerline joint. Stiffness analyses are performed for each unit load for each model.

Joint displacements for applied unit loads are formulated into flexibility matrices. These are inverted to obtain support stiffness matrices which were included in the reactor coolant loop model.

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INSERT C (Modification to 3.9(N).1.4d)

The reactor vessel supports are modeled using the WECAN computer program. Structure geometry, topology and member properties are used in modeling. Steam generator and reactor coolant pump supports are modeled as linear or non-linear springs.

For each operating condition, the loads (obtained from the RCL analysis) acting on the support structures are appropriately combined. Reactor coolant loop normal and upset conditions thermal expansion loads are treated as primary loading for the primary component supports. The adequacy of each member of the steam generator supports, reactor coolant pump supports, and piping restraints is verified by solving the ASME III subsection NF stress and interaction equations. The adequacy of the RPV Support Structure is verfied using the WECAN computer program and comparing the resultant stresses to the criteria given in ASME III subsection NF.

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Loads acting on the supports obtained from the reactor coolant loop analysis, support structure member properties, and influence coefficients at each end of each member are input into the WESAN program.

For each support case used, the following is performed:

1. Combine the various types of support plane loads to obtain operating condition loads (Normal, Upset, Emergency or Faulted).

2. Multiply member influence coefficients by operating condition loads to obtain all member internal forces and moments.

3. Solve appropriate stress or interaction equations for the specified operating condition. Maximum normal stress, shear stress, and combined load interaction equation values are printed as a ratio of maximum actual values divided by limiting values. ASME Boiler and Pressure Vessel Code Section III, Subsection NF, stress and interaction equations are used with limits for the operating condition specified.

~~The reactor vessel support structure is analyzed for all loading conditions using a finite element model. Vertical and horizontal forces delivered to the support structures from the reactor vessel shoe are applied to the structure, and element stresses and concrete forces obtained.~~

e. Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop include the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is ANS Safety Class 1 and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9(N)-2. The equipment is analyzed for: 1) the normal loads of deadweight, pressure and thermal, 2) mechanical transients of OBE, SSE, and pipe ruptures, and 3) pressure and temperature transients outlined in Subsection 3.9(N).1.1. INSERT D

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analysis, a set of loads are determined which should be larger than those seen in any single plant analysis. The "umbrella" loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses

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INSERT D (Modification to 3.9(N).1.4e)

including the effects of asymmetric subcompartment pressurization (for vessel nozzle breaks)

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of the components. Any deviations where the actual load is larger than the "umbrella" load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator and pressurizer are performed using 2 percent damping for the OBE and 4 percent damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed using the damping for bolted steel structures, that is, 4 percent for the OBE and 7 percent for the SSE (2 percent for OBE and 4 percent for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump since the main flange, motor stand, and motor are all bolted assemblies (see Section 5.4). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the lines connect to the primary system piping are orificed to a 3/8 inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

~~f. Reactor Vessel Support LOCA Loads~~

~~The LOCA analysis which is performed for the reactor vessel support loads includes non-axisymmetric pressure distributions on the internals and on the vessel exterior walls.~~

~~A detailed dynamic model of the reactor vessel and internals is prepared which includes the stiffnesses of the reactor vessel support and the attached piping. Hydraulic forces are developed in the internals for the break at the reactor vessel nozzle; these forces are characterized by time dependent forcing functions on the vessel and core barrel. In the derivation of these forcing functions, the fluid-structure (or hydroelastic) interaction in the downcomer region between the barrel and the vessel is taken into account. The break at the vessel nozzle also allows an asymmetric pressure distribution and a subsequent force on the side of the vessel is calculated on a time history basis for these asymmetric loads. As a result of the pipe break, loop mechanical loads are also applied to the vessel.~~

See new section 3.9N.1.5 on next page

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~~The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of the ASME Code, Section III. These valves are identified in Section 3.9N.3.2.~~

~~Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the line connect to the primary system piping are orificed to a 3/8 inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.~~

3.9N.1.5 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss of Coolant Accident

1. Introduction

This section presents the method of computing the reactor pressure vessel response to a postulated loss of coolant accident (LOCA). The structural analysis considers simultaneous application of the time-history loads on the reactor vessel resulting from the reactor coolant loop mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization (for postulated breaks in the reactor coolant pipe at the vessel nozzles). The vessel is restrained by reactor vessel support pads and shoes beneath four of the reactor vessel nozzles and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

Pipe displacement restraints installed in the primary shield wall limit the break opening area of the vessel nozzle pipe breaks to less than 144 square inches. This break area was determined to be an upper bound by using worst case vessel and pipe relative motions based on similar plant analyses. Detailed studies have shown that pipe breaks at the hot or cold leg reactor vessel nozzles, even with a limited break area, would give the highest reactor vessel support loads and the highest vessel displacements, primarily due to the influence of reactor cavity pressurization. By considering these breaks, the most severe reactor vessel support loads are determined. For completeness, an additional break outside the shield wall, for which there is no cavity pressurization, was also analyzed, specifically, the pump outlet nozzle pipe break.

2. Interface Information

Asymmetric reactor cavity pressurization loads were provided to Westinghouse by United Engineers and Constructors.

All other input information was developed within Westinghouse. This information includes: reactor internals properties, loop mechanical loads and loop stiffness, internal hydraulic pressure transients, and reactor support stiffnesses. These inputs allowed formulation of the mathematical models and performance of the analyses, as will be described.

3. Loading Conditions

Following a postulated pipe rupture at the reactor vessel nozzle, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of three phenomena: (1) reactor coolant loop mechanical loads, (2) reactor cavity pressurization forces and (3) reactor internal hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. This analysis is described in Section 3.9N.1.4C. The loop mechanical forces which are released at the broken nozzle are applied to the vessel in the RPV blowdown analysis.

Reactor cavity pressurization forces arise for the pipe breaks at the vessel nozzles from the steam and water which is released into the reactor cavity through the annulus around the broken pipe. The reactor cavity is pressurized asymmetrically with higher pressure on the side of the broken pipe resulting in horizontal forces applied to the reactor vessel. Smaller vertical forces arising from pressure on the bottom of the vessel and the vessel flanges are also applied to the reactor vessel. The cavity pressure analysis is described in Section 6.2.

The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a vessel inlet nozzle break and pump outlet nozzle break, the depressurization wave path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down and around the downcomer annulus and up through the fuel. In the case of an RPV outlet nozzle break the wave passes through the RPV outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus, for an outlet nozzle break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the inlet break. For both the inlet and outlet nozzle breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the

(210.44)

pressure transients. This phenomena has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708[9].

4. Reactor Vessel and Internals Modeling

The reactor vessel is restrained by two mechanisms: (1) the four attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and (2) four reactor vessel supports, two beneath reactor vessel inlet nozzles and two beneath reactor vessel outlet nozzles. The reactor vessel supports are described in Section 5.4.14 and are shown in Figures 5.4-12, and 3.8-17. The support shoe provides restraint in the horizontal directions and for downward reactor vessel motion.

The reactor vessel model consists of two non-linear elastic models connected at a common node. One model represents the dynamic vertical characteristics of the vessel and its internals, and the other model represents the translational and rotational characteristics of the structure. These two models are combined in the DARI-WOSTAS code[1] to represent motion of the reactor vessel and its internals in the plane of the vessel centerline and the broken pipe centerline.

The model for horizontal motion is shown in Figure 3.9N-12. Each node has one translational and one rotational degree of freedom in the vertical plane containing the centerline of the nozzle attached to the broken pipe and the centerline of the vessel. A combination of beam elements and concentrated masses are used to represent the components

(210.44)

including the vessel, core barrel, neutron panels, fuel assemblies, and upper support columns. Connections between the various components are either pin-pin rigid links, translational impact springs with damping or rotational springs.

The model for vertical motion is shown in Figure 3.9N-13. Each mass node has one translational degree of freedom. The structure is represented by concentrated masses, springs, dampers, gaps, and frictional elements. The model includes the core barrel, lower support columns, bottom nozzles, fuel rods, top nozzles, upper support structure, and reactor vessel.

The horizontal and vertical models are coupled at the elevation of the primary nozzle centerlines. Node 1 of the horizontal model is coupled with node 2 of the vertical model at the reactor vessel nozzle elevation. This coupled node has external restraints characterized by a 3 x 3 matrix which represents the reactor coolant loop stiffness characteristics, by linear horizontal springs which describe the tangential resistance of the supports, and by individual non-linear vertical stiffness elements which provide downward restaint only. The supports as represented in the horizontal and vertical models (Figures 3.9N-12 and 3.9N-13) are not indicative of the complexity of the support system used in the analysis. The individual supports are located at the actual support pad locations and accurately represent the independent non-linear behavior of each support.

5. Analytical Methods

The time-history effects of the cavity pressurization loads, internals loads and loops mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the

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displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time-history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

6. Results of the Analysis

As described, the reactor vessel and internals were analyzed for three postulated break locations. Table 3.9N-12 summarizes the displacements and rotations of and about a point representing the intersection of the centerline of the nozzle attached to the leg in which the break was postulated to occur and the vertical centerline of the reactor vessel. Positive vertical displacement is up and positive horizontal displacement is away from and along the centerline of the vessel nozzle in the loop in which the break was postulated to occur. These displacements were calculated using an assumed break opening area for the postulated pipe ruptures at the vessel nozzles of 144 in^2 and a double-ended rupture at the pump outlet nozzle. These areas are estimated prior to performing the analysis. Following the reactor coolant system structural analysis, the relative motions of the broken pipe ends are obtained from the reactor coolant loop blowdown analysis. The actual break opening area is then verified to be less than the estimated area used in the analysis and assures that the analysis is conservative.

The maximum loads induced in the vessel supports are due to the postulated pipe break. [redacted]. These loads are per vessel support and are applied at the vessel nozzle pad. It is conservatively assumed that the maximum horizontal and vertical loads occur

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simultaneously and on the same support, even though the time-history results show that these loads occur neither simultaneously nor on the same support. The largest vertical loads are produced on the support opposite the broken nozzle. The largest horizontal loads are produced on the supports which are perpendicular to the broken nozzle horizontal centerline. Note that the peak loads are conservative values since the break opening area for the vessel inlet nozzle break (as obtained from the dynamic loop analysis) is actually less than the estimated 144 square inch area used to generate the applied loads. If additional analysis was performed using the lower break opening area, the loads would be considerably reduced.

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The loads from these three sources, the internals reactions, reactor cavity pressure loads, and the loop mechanical forces, are applied simultaneously in a nonlinear elastic dynamic time history analysis on the model of the vessel, supports and internals. The results of this analysis are the dynamic loads on the reactor vessel supports and vessel time history displacements. The maximum loads are combined with other applicable loads, such as seismic and deadweight and applied statically to the vessel support structure. The maximum stresses in the support are calculated and compared to faulted condition stress allowables given in Subsection 3.9(N).1.4g.

3.9(N).1.6 Stress Criteria for Class 1 Components and Component Supports Type all Caps

All Class 1 components and supports are designed and analyzed for the Design, Normal, Upset, and Emergency Conditions to the rules and requirements of the ASME Code Section III. The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of Faulted Conditions are those that are defined in Appendix F of the ASME Code with supplementary options outlined below:

1. Elastic System Analysis and Component Inelastic Analysis

   This is an acceptable method of evaluation for Faulted Conditions if the rules of F1323.1(a) are met for component supports, within the scope of Subsection NF and if primary stress limits for components are taken as greater of 0.70 S_u or $S_y + 1/3\ (S_u - S_y)$ for membrane stress and greater of 0.70 S_{ut} or $S_y + 1/3\ (S_{ut} - S_y)$ for membrane-plus-bending stress, where material properties are taken at appropriate temperature.

   If plastic component analysis is used with elastic system analysis or with plastic system analysis, the deformations and displacements of the individual system members will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

2. Elastic/Inelastic System Analysis and Component/Test Load Method

   The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

   If the component/test load method is used with elastic or plastic system analysis, the deformations and displacements of the individual component members taken from the test load

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method data at the loads resulting from the system analysis will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

3. Component Support Buckling Allowable Load

In the design of component supports, members compressive axial loads are limited to 0.67 times the critical buckling strength. If, as a result of more detailed evaluation of the supports the member compressive axial loads can be shown to safely exceed 0.67 times the critical buckling strength for the faulted condition, verification of the support functional adequacy will be documented and submitted to the NRC for review. The member compressive axial loads will not exceed 0.67 times the critical buckling strength without NRC acceptance. In no case will the compressive load exceed 0.9 times the critical buckling strength.

Loading combinations and allowable stresses for ASME III Class 1 components and supports are given in Tables 3.9(N)-2 and 3.9(N)-3. For Faulted condition evaluations, the effects of the safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) are combined using the square-root-of-the-sum-of-the-squares (SRSS) method. Justification for this method of load combinations is contained in References (4) and (5).

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3.9(N).2 Dynamic Testing and Analysis

3.9(N).2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

A preoperational piping vibrational and dynamics effects testing program will be conducted for the reactor coolant loop/supports system during preoperational testing. The purpose of these tests will be to confirm that the system has been adequately designed and supported for vibration, as required by Section III of the ASME Code, paragraph NB-3622.3. The tests will include reactor coolant pump starts and trips. If vibration is experienced, which, from visual observation, appears to be excessive, either: 1) an instrumented test program on the piping, will be conducted and the system reanalyzed to demonstrate that the observed levels will not cause ASME Code stress and fatigue limits to be exceeded, 2) the cause of the excessive vibration will be eliminated, or 3) the support system will be modified to reduce the vibration. Particular attention will be provided at those locations where the vibration is expected to be the most severe for the particular transient condition being studied.

It should be noted that the layout, size, etc., of the reactor coolant loop and surge line piping used in the Seabrook plants is very similar to that employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant

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INSERT E (Modification to 3.9(N).1.6 and 3.9(N).1.7)

For faulted conditions analysis of class 1 branch piping attached to the reactor coolant loop, Equation (9) of ASME III subsection NB-3652 is applied with a stress limit of $3.0S_m$. This criterion provides sufficient assurance that the piping will not collapse or experience gross distortion such that the function of the system would be impaired. The basis for this position is described in Westinghouse response to NRC Question 110.34 on the RESAR-414 application (Docket no. STN 50-572), which subsequently received a preliminary design approval (PDA) in Nov., 1978.

3.9(N).1.7 Analytical Methods for RCS Class 1 Branch Lines

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and dynamic structural analysis for the effect of a reactor coolant loop pipe break.

The integrated Class 1 piping/supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the reactor coolant loop, and the stiffness of supports which affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The Class 1 piping system models are constructed for the WESTDYN computer program, which numerically describes the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads are also computed by mutiplying the stiffness matrix by the displacement vector at the support point.

Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system which will appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, reactor coolant pump, and pressurizer, on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors which resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of modal superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

Loss of Coolant Accident

The mathematical models used in the seismic analyses of the Class 1 lines are also used for RCL pipe break effect analysis. To obtain the proper dynamic solution both for lines attached to the unbroken loops and lines attached to the broken loop, the time history deflections from the analysis of the reactor coolant loop are applied at branch nozzle connections.

Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in Section 3.9.1.1 are considered in the fatigue evaluation.

The thermal quantities T_1, $\bar{T}_2$, and a_aT_a, $-a_bT_b$ are calculated on a time history basis, using a one-dimensional finite difference heat transfer computer program. Stresses due to these quantities were calculated for each time increment using the methods of NB-3650 of ASME III.

For each thermal transient, two loadsets are defined, representing the maximum and minimum stress states for that transient.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic monments at each point, thus insuring the most conservative combinations of seismic loads are used in the stress evaluation.

The WESTDYN computer program is used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors and cumulative usage factors for all possible load

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set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cylces $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

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c. Dynamic Analysis

The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses on the CRDS.

d. Control Rod Drive Mechanisms

The control rod drive mechanism (CRDM) pressure housings are Class I components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDM's when a seismic disturbance has been postulated, to confirm the ability of the pressure housing to meet ASME Code, Section III, allowable stresses and to evaluate the effect of the seismic event on the drop time.

INSERT F

Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the CRDM's are:

1. 5/8 inch step,
2. 147 inch travel,
3. 360 pound maximum load,
4. Step in or out at 45 inches/minute (72 steps/minute),
5. Electrical power interruption shall initiate release of drive rod assembly,
6. Trip delay time of less than 150 milliseconds - free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption, no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F,
7. 40 year design life with normal refurbishment.

3.9(N).4.3 Design Loads, Stress Limits, and Allowable Deformations

a. Pressure Vessel

The pressure retaining components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The

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INSERT F (Modification to 3.9(N).4.2)

The Control Rod Drive Mechanisms (CRDM's) are evaluated for the effects of postulated reactor vessel inlet nozzle and outlet nozzle limited displacement breaks. A time history analysis of the CRDM's is performed for the vessel motion discussed in Section 3.9(N).1.5. A model of the CRDM's is formulated with gaps at the upper CRDM support modeled as nonlinear elements. The CRDM's are represented by beam elements with lumped masses. The translation and rotation of the vessel head is applied to this model. The resulting loads and stresses are compared to allowables to verify the adequacy of the system.

(21044)

(c) Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing were selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F, the inside diameter of the coil stack is 7.308/7.298 inches, and the outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism, due to operating temperature of the CRDM, results in the minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing outside diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above, it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

To verify the acceptability of the above tolerances, four coil stack assemblies were removed from four hot control rod drive mechanisms, mounted on 11.035 inch centers on a 550°F test loop, allowed to cool, and then placed without incident.

(d) Coil Fit in Coil Housing

CRDM and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot control rod drive mechanism.

INSERT G

3.9(N).4.4 CRDS Performance Assurance Program

a. Evaluation of Material's Adequacy

The ability of the pressure housing components to perform throughout the design lifetime, as defined in the equipment specification, is confirmed by the stress analysis report required by the ASME Code, Section III.

Internal components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment, as confirmed by life tests (Reference 12). Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single control rod drive mechanism.

To confirm the mechanical adequacy of the fuel assembly, the control rod drive mechanism and rod cluster control assembly, functional

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INSERT G (Modification to 3.9(N).4.3)

3.9(N).4.3.1 Evaluation of Control Rod Drive Mechansims and Supports

The control rod drive mechanisms (CRDM's) and CRDM support structure are evaluated for the loading combinations outlined in Table 3.9(N)-2.

A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. These include RPI plate impact, tie rods, and lifting leg clevis/RPV head interface. The time history motion of the reactor vessel head, obtained from the RPV analysis is input to the dynamic model. Maximum forces and moments in the CRDMs and support structure are then determined. For the seismic analysis, the structrual model is linearized and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDMs for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evalutation of the CRDMs. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

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test programs have been conducted on a full-scale 12 foot control rod. The 12 foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test, the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the CRDM meets the design requirement of 2.2 seconds from start of rod cluster control assembly motion to dashpot entry. This trip time requirement will be confirmed for each control rod drive mechanism prior to initial reactor operation and at periodic intervals after initial reactor operation, as required by the proposed Technical Specifications.

These tests have been reported in Reference (12).

INSERT H

There are no significant differences between the prototype control rod drive mechanisms and the production units. Design materials, tolerances and fabrication techniques are the same (see Section 4.5).

It is expected that all control rod drive mechanisms will meet specified operating requirements for the duration of plant life, with normal refurbishment. Latch assembly inspection is recommended after 12.5×10^6 steps have been accumulated on a single CRDM.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited to one as discussed in the proposed Technical Specifications.

In order to demonstrate proper operation of the control rod drive mechanism, and to ensure acceptable core power distributions during rod cluster control assembly partial-movement, checks are performed on the rod cluster control assemblies (refer to Technical Specifications). In addition, periodic drop tests of the rod cluster control assemblies are performed at each refueling shutdown, to demonstrate continued ability to meet trip time requirements.

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INSERT H (Modification to 3.9(N).4.4)

In addition, dynamic testing programs have been conducted by Westinghouse and Westinghouse Licensees to demonstrate that control rod scram time is not adversely affected by postulated seismic events. Acceptable scram performance is assured by also including the effects of the allowable displacements of the driveline components in the evaluation of the test results.

Revised

RAI 210.45 (3.9(N).2.5, Page 3.9(N)-41)

Your statement that the loading imposed by the SSE is generally small compared to blowdown loadings implies that in certain cases you have neglected loads due to an SSE. If this is true, provide analysis details justifying your doing so.

RESPONSE:

The loading imposed on the reactor internals by the Safe Shutdown Earthquake (SSE) for the Seabrook plant is small compared to the blowdown loading. As stated on FSAR Page 3.9(N)-43, "The stresses due to the Safe Shutdown Earthquake (vertical and horizontal components) were combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection". The FSAR ~~will be~~ is revised to clarify the subject statement.

3.9(N).2.5

in Amendment 47

(210.45)

component ~~may~~ ARE be additive in certain cases and, therefore, the combined loading ~~must be~~ considered~~. In general, however,~~ EVEN THOUGH the loading imposed by the earthquake is GENERALLY small compared to the blowdown loading.

The summary of the mechanical analysis follows:

a. Vertical Excitation Model for Blowdown

For the vertical excitation, the reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. Also incorporated in the multi-mass system is a representation of the motion of the fuel elements relative to the fuel assembly grids. The fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers. Coulomb-type friction is assumed in the event that sliding between the rods and the grid fingers occurs. In order to obtain an accurate simulation of the reactor internals response, the effects of internal damping, clearances between various internals, snubbing action caused by solid impact, Coulomb friction induced by fuel rod motion relative to the grids, and preloads in hold down springs have been incorporated in the analytical model. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses, while elastic elements are represented by springs.

The appropriate dynamic differential equations for the multi-mass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program which computes the response of the multi-mass model when excited by a set of time-dependent forcing functions. The appropriate forcing functions are applied simultaneously and independently to each of the masses in the system. The results from the program give the forces, displacements and deflections as functions of time for all the reactor internals components (lumped masses). Reactor internals response to both hot and cold leg pipe ruptures are analyzed.

b. Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

c. Core Barrel

For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

Revised

RAI 210.46 (3.9(N).4.3, Page 3.9(N)-60)

The statement "The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials" needs clarification. What are these stress limits and from what source were they obtained?

RESPONSE:

The stress limits associated with the design of the Control Rod Drive System are those defined in Section III of the ASME Code. The subject statement in the FSAR is only intended to reiterate the basic intent of the ASME Code. For the Control Rod Drive System, ASME Code limits have been satisfied where required.

Revised

RAI 210.47 (3.9(N).4.3, Page 3.9(N)-61)

Provide assurance that deformation limits are sufficient to guarantee control rod drive system integrity and functioning after a dynamic event such as an OBE.

RESPONSE:

The control rod drive system (CRDS) integrity (deformation) during an OBE is assured by limiting the allowable stress levels for the pressure retaining components and reactor internals to those defined by Subsections NB and NG, respectively, of Section III of the ASME Code. Thus, after the OBE, the geometrical relationship between the various components of the CRDS is basically the same as the pre-OBE configuration. It is emphasized that both the stress and deformation limits are considered in the evaluation of the CRDS and reactor internals to ensure the integrity of the CRDS and insertability of the control rods.

Revised

RAI 210.48 (3.9(N).5.2, Page 69)

The statement "The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the material" needs clarification. What are these stress limits and from what source were they obtained?

RESPONSE:

The stress limits associated with design of the reactor internals are those defined in NG-3000 of Section III of the ASME Code. As identified in ~~Item~~ RAI 210.46 the above statement is intended to reiterate the basic intent of the ASME Code. Additionally, as identified in ~~Item~~ RAI 210.49, the extent of compliance with the ASME Code will be included in an FSAR change.

Revised

RAI 210.49 (3.9(N).5.2, Page 3.9(N)-68 to 71)

Subsection NG, ASME Code Section III should be referenced as the design criteria for all design analyses, not just for the design basis accident.

RESPONSE:

The design and fabrication of the Seabrook core support structures conform to the requirements of the Subsection NG of Section III of the ASME Code. By contract, this plant preceeded the application of Subsection NG and, therefore, these internals are not "Code Stamped" and no specific Code stress report is required. The Seabrook plant reactor internals are identical in nature to the SNUPPS reactor internals which are stamped and documented to code requirements. The Seabrook FSAR ~~will be~~ revised to reflect the above stated comparison with ASME Code requirements.

Subsection 3.9(N).5.3 has been

(210.49)

ATTACHMENT TO ITEM 49

l. One or more loops out of service

m. All operational transients listed in Table 3.9(N)-1

n. Pump overspeed

o. Seismic loads (Operating Basis Earthquake and Safe Shutdown Earthquake)

p. Blowdown forces (due to cold and hot leg break).

The main objective of the design analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but to also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analysis on the reactor internals is provided in Section 3.9(N).2.

As part of the evaluation of design loading conditions, extensive testing and inspections are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections, are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

3.9(N).5.3 Design Loading Categories

The combination of design loadings fit into either the normal, ~~upset, emergency~~ or faulted conditions, as defined in the ASME Code, Section III INSERT *

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions, as summarized in Table 3.9(N)-1.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

3.9(N).5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

FSAR PG 3.9(N)-69

INSERT A for Response to Question 210.49

However, it should be noted that by contract the reactor internals for the Seabrook plant preceed the applicability of subsection NG of the ASME Code. Therefore, these internals are not "Code Stamped" and no specific stress report is required. Nevertheless, these reactor internals are designed to meet the intent of subsection NG of the ASME Code.

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New Response at 5/11/82 mtg.

RAI 210.50 (3.9(N).5.4, Page 3.9(N)-70)

Verify that reactor internals are designed in accordance with Standard Review Plan 3.9.3 "Core Support Structures" or justify alternate design criteria.

RESPONSE:

1. Design and service loading conditions for core support structures are given in Subsection 3.9(N).5.2.

2. The combination of design and service loadings fit into either the normal and upset, emergency or faulted conditions, as defined in Subsection NG of Section III of the ASME Code.

3. The stress limits associated with the design of the core support structures are those defined in Subsection NG (NG-3000) of Section III of the ASME Code.

4. The deformation criteria for reactor internals components and core support structures is established in regard to mechanical integrity such that adequate core cooling and core shutdown must be assured. The deformation limits for reactor internals and core support structures are given in Table 3.9(N)-12.

5. Dynamic system analysis of the reactor internals for faulted conditions is performed as discussed in Subsection 3.9(N).2.5.

The above information demonstrates that the reactor internals design and analysis are consistent with Section 3.9.3 of the Standard Review Plan for Core Support Structures.

Revised

RAI 210.51 (3.9(N).5.4, Page 3.9(N)-71)

What are the stresses associated with the maximum deflections in Table 3.9(N)-17? State the basis for these deflection limits. Justify the lack of safety margin for radial outward deflection.

RESPONSE:

Stresses corresponding to the maximum deflections given in Table 3.9(N)-12 are not calculated because these deflections characterize certain limits (called no-loss-of-function) so that the operability of the reactor is not impeded (e.g., control rod insertability is assured). For example, the radial inward deflection of 8.2 inches describes the no-loss-of-function of upper barrel during hot leg break of LOCA. This implies that during hot leg break, the radial inward deflection of the upper barrel be limited to 8.2 inches so as not to impair the operation of (deflect or bend) guide tube assemblies. The 1.0 inch radial outward deflection of the upper barrel during cold leg break defines the limit in front of the unbroken nozzles so as not to impair the efficiency of the Emergency Core Cooling System.

For the upper package, the no-loss-of-function deflection 0.15 inch refers to the vertical upward deformation of the guide tubes. This limit is set to preclude any axial compressive buckling loads on the guide tubes. Finally, 1.75 inch no-loss-of-function deflection defines the limit on the transverse displacement of the guide tubes so as not to impair the control rod insertion function.

Therefore, the basis for these deflection limits is that an adequate core cooling and core shut down is assured. The fact that the calculated deflections are less than the allowable displacements, provides additional conservatisms on the design of reactor internals.

Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections in addition to a stress criteria to assure integrity of the components.

Although the Seabrook reactor internals are not contractually required to meet the ASME Code as identified in ~~Item~~ 49, the code criteria for faulted conditions (Appendix F) is used as design basis for evaluating acceptability of calculated stresses, and the resulting stresses and deformations are below the established limits.

RAI 210.49

Revised

RAI 210.52 (3.9(N).2.3, Table 3.9(N)-1)

Provide information on how the number of occurrences of steady state fluctuations were determined. How was the effect of transients listed in this table considered for BOP equipment?

RESPONSE:

Page 3.9(N)-5 of the FSAR provides a description of the steady state fluctuations.

Initial Steady State Fluctuations

Initial steady state fluctuations result from conditions of small moderator and doppler coefficients and high rod worths. Control rod cycling can occur causing temperature cycles of $\pm$ 3^{o}F. The corresponding RCS pressure fluctuations are limited to $\pm$ 25 psi by pressurizer spray and backup heaters. Using maximum rod worth data from operating plants, the cycling rate was found to be about one step per minute, giving a continuous cycling period of about two minutes. Because of fuel burnup and an increasingly more negative moderator coefficient, the cycling period increases and no cycling occurs beyond the first two fuel cycles (20 full power months). The total cumulative number of cycles is approximately 150,000.

Random Steady State Fluctuations

Random RCS temperature and pressure fluctuations were obtained from operating plant data. Maximum fluctuations of $\pm$ 0.5^{o}F and $\pm$ 6 psi were found to occur over a period of approximately six minutes. Assuming plant availability to be 85%, this would result in approximately $3x10^6$ total fluctuations during the 40 year design life of the plant.

The determination of the number of steady state fluctuations is therefore based on NSSS design and operating experience.

Steady state fluctuations (initial and random) have no effect on BOP equipment. However, the remainder of the RCS design transients listed in Table 3.9(N)-1 may impact BOP equipment. Determination of such impact, if any, falls within the scope of UE&C. Interface criteria are transmitted to the utility and/or UE&C in the form of internal Westinghouse design documents. This information may be used by the BOP designer on an as applicable basis. ~~No additional information is required from Westinghouse.~~

RAI 210.53 (3.9.6.2)

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.

RESPONSE:

A valve test program is in preparation and, when completed, will address inservice testing of valves whose function is to perform pressure isolation between high pressure reactor coolant and low pressure systems. Specific information regarding valve testing criteria, frequency, and exceptions will be made available to the NRC Project Manager for review by January 1, 1983. This date is consistent with the previously stated submittal date for information relative to the Inservice Inspection Program.

RAI 210.54

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.

RESPONSE:

Subsection 3.9(B).2, Dynamic Testing and Analysis, is presently undergoing an extensive review to define scope and programatic requirements. An update of this section, in conjunction with applicable portions of Chapter 14, will be provided to the NRC by October, 1982.

RAI 210.55

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 ve verified as follows:

a. During initial system heatup and cooldown, at specified temperature intervals for any system which attains opperating 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.

Revised

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.

RESPONSE:

The response to this RAI was included in our November 27, 198~~2~~4, submittal in response to the Acceptance Review RAIs, and was subsequently incorporated into FSAR Amendment 44 and Amendment 45 (Subsection 3.9(B).3.4(d)).

RAI 210.56 (3.2.1, Table 3.2-2, Sheet 3 of 31)

Explain the use of ASME VIII for the design of the pressurizer discharge piping.

RESPONSE:

Refer to revised Sheet 3 of Table 3.2-2, which ~~will be~~ was provided in Amendment 45. This was a typographical error.

RAI 210.57 (3.6(B).2.1, Page 3.6(B)-7)

Explain in more detail how intermediate break points are determined in Class 1 BOP piping when stresses and usage factors are below 2.4 Sm on 0.1, respectively.

RESPONSE:

BOP

The criteria of Regulatory Guide 1.46 were followed for the intermediate break point determinations. FSAR Paragraph 3.6(B).2.1a ~~is being~~ was revised in Amendment 45.

| Revised

NSSS

For Class 1 analysis outside the RCS, Westinghouse provides stress and usage factors to UE&C. UE&C then takes the analysis results and determines break locations.

| New Response at 5/11/82 mtg.

210.58 (3.6(B).2.1, Page 3.6(B)-7)

It is the staff's position that if a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure irrespective of the fact that the pipe break criteria might not require such a break location to be postulated. Provide assurance that the Seabrook plant meets the above requirement.

RESPONSE: In evaluating the effects of high energy line breaks on essential components, it was found that protection is most often provided by separation distance due to the high energy lines being located far enough away from essential components that the danger of impact or jet impingement did not exist. In the case of the electrical trays in the control building, which were separated by the building wall from the adjacent main steam and feedwater lines, a guard pipe was provided to prevent jet impingement from the main steam line, and an energy-absorbing bumper was provided to prevent impact from main steam or feedwater lines that could damage the wall.

RAI 210.59 (Appendix 3C, Page 4)

For circumferential breaks that are axially restrained, provide justification that the axial and lateral movement of the pipe will result in the fan jet impingement that is assumed. The staff's position is that a fan jet can occur when the broken piping is physically restrained from significant separation (axial pipe movement equal to or less than 1/2 pipe diameter and lateral pipe movement less than pipe wall thickness).

RESPONSE:

Seabrook BOP piping is not axially restrained so as to restrict pipe motion in such a fashion as to form a fan jet. Fan jet calculations were not used in the analysis.

RAI 210.60

What is the maximum allowable tip deflection of a restrained whipping pipe? Provide assurance that the deflections of the restrained whipping pipe will not affect the function of any safety-related components.

RESPONSE:

BOP

For BOP piping, pipe whip restraints are provided to maintain the motion of the ruptured pipe end within controlled limits. The limit of motion is the area within which no essential component can be affected by impact or jet impingement.

NSSS

Westinghouse designs pipe whip restraints for the RCS to eliminate any postulated pipe whip concerns. Based on the Westinghouse restraint design, the maximum allowable tip deflection of a restrained whipping pipe is generally limited to the thickness of the pipe. This criterion precludes the whipping pipe affecting the function of any safety related component.

New Response at 5/11/82 mtg.

RAI 210.61

Provide a description on examples of the different types of pipe whip restraints and jet impingement barriers that are used in the plant.

RESPONSE:

BOP

For BOP piping, pipe whip restraints are either bumpers (with or without energy-absorbing crushable pads), guides, U-bolt restraints, or a combination of these. Guides are structural steel frames surrounding the pipe. U-bolts are only used in tension. Jet impingement barriers consist of sleeves or guard pipes. Revised

NSSS

The NSSS supplier only provides pipe whip restraints for the RCS. These restraints are described in FSAR Chapter 5. New Respons

Revised

RAI 210.62 (3.6(N).2.3, Page 3.6(N)-9)

When calculating the dynamic effects of jet impingement, what values are assumed for Ko for the various initial fluid conditions?

RESPONSE:

When performing jet impingement analysis for the RCS, Westinghouse uses a K_o value of 1.3 for hot leg breaks and a K_o value of 1.57 for cold leg breaks. These values are based on ANS standard 58.2. The FSAR ~~will be~~ was revised in Amendment 47 to incorporate a reference to ANS 58.2.

(210.62)

ATTACHMENT TO ITEM 62

The shape of the target affects the amount of momentum change in the jet, and thus affects the impingement force on the target. The target shape factor is used to account for target shape which do not deflect the flow 90 degrees away from the jet axis.

The method used to compute the jet impingement load on a target is one of the following:

1. The dynamic effect of jet impingement on the target structure is evaluated by applying a step load whose magnitude is given by

$$F_j = K_o P_o A_{mB} R S$$

where

F_j = jet impingement load on target

K_o = dimensionless jet thrust coefficient based on initial fluid conditions in broken loop (K_0 IS BASED ON ANS 58.2 (ANSI 176), "PROTECTION OF NUCLEAR POWER PLANTS AGAINST PIPE RUPTURE".

P_o = initial system pressure

A_{mB} = calculated maximum break flow area

R = fraction of jet intercepted by target

S = target shape factor

Discharge flow areas for limited flow area circumferential breaks are obtained from reactor coolant loop analyses performed to determine the axial and lateral displacements of the broken ends as a function of time. A_{mB} is the maximum break flow area occurring during the transient, and is calculated as the total surface area through which the fluid must pass to emerge from the broken pipe. Using geometrical formulations, this surface area is determined to be a function of the pipe separation (axial and transverse) and the dimensions of the pipe (inside and outside diameter).

If a simplified static analysis is performed instead of a dynamic analysis, the above jet load (F_j) is multiplied by a dynamic load factor. For an equivalent static analysis of the target structure, the jet impingement force is multiplied by a dynamic load factor of 2.0. This factor assumes the target can be represented as essentially a one degree of freedom system and the impingement force is conservatively applied as a step load.

Revised

RAI 210.63 (Table 3.6(N)-2)

Why are only seven break locations listed? There are eleven design breaks postulated. Provide the CUF and moments for the other four design break locations. In addition, provide any other modes where the CUF exceeds 0.1. Provide the CUF values specific for Seabrook at each design break location and at any other locations where the CUF exceeds 0.1.

RESPONSE:

Footnote a in Table 3.6(N)-2 addresses the other four pipe break locations postulated in the RCS. The fatigue analysis for the RCS is performed on a generic basis. The cummulative usage factors do not change from plant to plant. For the Seabrook plant, a specific moment analysis is performed to verify that the specific plant moments are enveloped by the generic analysis performed by Westinghouse and described in WCAP-8082.

New Response at 5/11/82 mtg

RAI 210.64 (3.7.3)

It is the staff's position that the use of the overlap technique in piping subsystem analysis is not generally acceptable. Provide a discussion on the use of the overlap technique with respect to NUREG/CR-1980. Identify those piping subsystems that use the overlap technique.

RESPONSE:

The overlap method is used primarily for those cases in which branch piping cannot be decoupled from the main run piping. When this situation arises, a sufficient subsection of the branch piping is included in the main run model to ensure inclusion of the branch effects. Conversely, subsequent branch analyses include adequate sections of the run piping.

The criterion employed requires that each decoupled analytical model includes a region of overlap common to both subsystems which contain the necessary supports to minimize translations as well as bending moments transmitted across overlapping boundaries. In practice, the decoupling is accomplished by extending the model to include supports which provide restraining actions in at least two x-x, two y-y and one z-z direction(s) (x-x, y-y, and z-z are the local piping coordinates with z-z being parallel to the piping centerline). Stresses and support loads for the run are those determined for the model elements representing it. Branch pipe stresses are determined in a similar manner. However, in either case, whether the run or the branch pipe is being analyzed, the pipe stresses in the overlapped or extension regions are checked against the appropriate allowables.

In those rare cases where the overlapping technique is used for analyzing large main run piping systems, the criterion discussed above for selecting supports and restraints in the overlap region which effectively decouple the subsystems is used. Here, results from overlapped subsystem models are treated in a conservative manner. Pipe stresses in the overlapped region are deemed to be the maximum found in either of the subsystem models, whereas total reaction loads for supports in these regions are computed by taking the absolute sum (for seismic loads) of the loads determined for each subsystem.

Direct comparisons of the preceding techniques with the test cases presented in NUREG/CR-1980 are difficult because of the rather simplified nature of those test cases. Nevertheless, UE&C's methods are considered adequate. The specific boundary selection criteria are accompanied by other conservatisms. These conservatisms are inherent in the applied response spectra. In fact, response spectra envelopes are applied when applicable. In addition, UE&C's support envelopes are applied when applicable. In addition, UE&C's support design philosophy dictates adequately stiff supports relative to the pipe sizes for which they are designed. These measures, in conjunction with the above modeling techniques, are considered to provide acceptable analytical methods and designs. Substantiating test cases comparing coupled and decoupled models using the techniques presented herein are being performed.

The piping systems which have been analyzed using the preceding methods are listed below:

CS - Chemical and Volume Control

CC - Primary Component Cooling Water

CBS - Containment Spray

RM - Residual Heat Removal

SW - Service Water

New Response at 5/11/82 meeting

RAI 210.65 (3.9(B).1.4, Page 3.9(B)-7)

The FSAR states that the load combination's and stress limits of Table 3.9(B)-7 provide assurance that essential Class 2 and 3 piping systems will retain their functional integrity. The staff does not agree that the use of 1.8 Sh stress limits will assure the piping functionality. We have provided the applicant with our acceptance criteria for functional capability (Reference: NRC evaluation of General Electric Topical Report NEDO-21985 dated September 1978). Provide assurance that the Seabrook plant meets our acceptance criteria for piping functional capability.

RESPONSE:

1. The allowable stress limit of 1.8 S_h for essential Class 2 and 3 piping systems is lower, in magnitude, than the functional capability limit of 1.5 S_y at temperature below 400°F for piping materials used. Therefore, functional capability is assured below 400°F.

2. At temperatures above 400°F, the stress limit (for Service Level C only) of 1.8 S_h is greater than 1.5 S_y for austenitic steels. The actual percentage difference between the allowable stress limit (1.8 S_h) and the functional capability limit (1.5 S_y) is small (6-8% depending upon material). NUREG/CR-0261, which is the basis for the NRC position on functional capability, states: "The C-Limits of 1.5 S_y involve an engineering judgment in which we take advantage of .... It is our judgment that, under C-Limits, straight pipe will not be subject to excessive strains nor will it lose a significant part of its flow area."[(1)] Since the criteria used for piping analysis have significant built-in conservatisms (e.g., magnitude of loads, method of determining "worst" stress condition, and the functional capability limit was chosen based to a large extent on judgment, we believe that the small difference between limits does not impair functional capability in this situation.

   In conclusion, we belive that the allowable limit of 1.8 S_h assures functional capability.

(1) Evaluation of the Plastic Characteristics of Piping Products in Relation to ASME Criteria, E. C. Rodabough, S. E. Moore. July 1978, NUREG/CR-0261, Page 59.

RAI 210.66

Provide a more detailed description of the loads, load combinations, and stress limits that are used in the design of ASME Code Class 1, 2 and 3 components and component supports.

RESPONSE:

BOP

The plant loading conditions and load combinations are summarized in Table 3.9(B)-2 for ASME components and their supports. Stress limits applicable to the various components (excluding supports) for each loading condition are summarized in Tables 3.9(B)-3 through 3.9(B)-5. Note (1) on each of these tables instructs the reader to refer to Table 3.9(B)-2 for definitions of the plant loading conditions.

Tables 3.9(B)-2 and 3.9(B)-3 ~~are being~~ have been revised in Amendment ~~45~~. 47.

Revised

Supports are addressed in FSAR Subsection 3.9(B).1.4(a). All component support designs, except those listed in the response to RAI 210.29, satisfy the design requirements and stress allowables of ASME III, Subsection NF. Those listed in the RAI 210.29 response are designed according to AISC criteria.

NSSS

For loading combinations and stress limits for Westinghouse ASME Code Class 1, 2, & 3 components and component supports, refer to tables in FSAR Section 3.9(N).

FSAR Subsection 3.9(N).3.4 will be revised to state compliance with the ASME Code Subsection NF for Class 2 and 3 components supports procured after July 1, 1974.

New Respon: at 5/11/82 meeti

(210.66)

ATTACHMENT TO ITEM 66

valve exercising and inspection to assure the functional ability of the valve.

The pressurizer safety valves are qualified by the following procedures: 1) stress and deformation analyses of critical items which may affect operability for faulted condition loads, 2) in shop hydrostatic and seat leakage tests, and 3) periodic in-situ valve inspection. In addition to these tests, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event.

Using these methods, active valves are qualified for operability during a faulted event. These methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary.

b. Pump Motor and Valve Operator Qualification

Active pump motors (including vital pump appurtenances) and active valve motor operators are seismically qualified in accordance with IEEE Standard 344-1975. The seismic qualification program for this electrical equipment is further described in Section 3.10(N) and the Equipment Qualification Data Packages referenced therein.

3.9(N).3.3 Mounting of Pressure Relief Devices

Refer to Subsection 3.9(B).3.3.

3.9(N).3.4 Component Supports (ASME Code Class 2 and 3)

See Subsection 3.9(N).1 for ASME Code Class 1 component supports.

a. Component Supports for Components Procured After July 1, 1974

Class 2 and 3 component supports are designed and analyzed for Design, Normal, Upset, Emergency, and Faulted conditions to the rules and requirements of Subsection NF of Section III of the ASME B&PV Code (1974 Edition). The design analyses or test methods and associated stress or load allowable limits that are used in the evaluation of linear supports for Faulted conditions are those defined in Appendix F of the ASME Code. ~~For linear type supports designed by analyses for ASME Code Class 2 and 3 components, the increased design limit for stress range identified in NF-[illegible] is limited to the smaller of 2 Sy or Su unless otherwise justified by shakedown analysis. Plate and shell type supports satisfy the~~

(210.66)

ATTACHMENT TO ITEM 66

~~following stress criteria for faulted conditions: $\sigma_1 \leq 2.0S$, $\sigma_1 + \sigma_2 \leq 2.4S$. ($\sigma_1$ and σ_2 are as defined in [illegible] ASME III, 1974 Edition).~~

b. <u>Component Supports for Components Procured Prior to July 1, 1974</u>

Class 2 and 3 supports are designed as follows:

1. <u>Standard Component Supports</u>

(a) Normal - The allowable stresses or load ratings of MSS-SP-58 are used.

(b) Upset - For upset conditions, the allowable stresses or load ratings are 20 percent higher than those specified for normal conditions.

(c) Emergency - For emergency conditions, the allowable stresses or load ratings are 80 percent higher than those specified for normal conditions. Supports (rod hangers and struts) are checked for elastic stability when applicable, and maximum compressive load does not exceed critical buckling load as specified by the applicable codes and design standards.

(d) Faulted - The allowable stresses or load ratings of MSS-SP-58 are based on a factor of safety which is greater than or equal to four, i.e., the allowble stress is less than or equal to one-fourth the minimum tensile stress of the material. The allowable stresses for faulted conditions are thus less than or equal to 0.6 times the minimum tensile stress of the material, i.e., 2.4 times one-fourth the minimum tensile stress of the material equals 0.6 times the minimum tensile stress. This low allowable stress (associated factor of safety equals 1.67) is adequate to assure that active componentsare properly supported for faulted conditions.

2. <u>Linear Type Supports</u>

(a) Normal - The allowable stresses of AISC-69 Part 1 are employed for normal condition limits.

(b) Upset - Stress limits for upset conditions are 33 percent higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC-69 Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

(c) Emergency - Not applicable.

RAI 210.67

Provide the stress limits used for NF bolts.

RESPONSE:

BOP

1. SA-325 Bolts:

   1/2" to 1"Ø bolts - Sy = 92 ksi
   Su = 120 ksi

   a. Friction Type Joint:

      Allowable slip resistance per ASME XVII - 2461.4

      Ps = mnTiKs

      Ps = 21 x As kips/bolt

      where:

      Slip coeff. Ks = 0.25

      Initial clamping force

      Ti = 0.7 x 120 x As = 84 As kips

      As = tensile stress area of the bolt, sq. in.

      Number of shear planes per bolt = m = 1

      Number of bolts = n = 1

   b. Bearing type joint:

      Allowable tensile stress = $\frac{Su}{2}$ = 60 ksi

      Allowable shear stress = $\frac{0.62Su}{3}$ = 24.8 ksi

2. SA-307 Bolts

   1/2" Ø to 1"Ø Bolts - Sy = 36 ksi
   Su = 58 ksi

   Allowable tensile stress = $\frac{Su}{2}$ = 29 ksi

   Allowable shear stress = $\frac{0.62Su}{3}$ = 12 ksi

Revised

(1-2 ...)

NSSS

In the design of bolting for ASME Code Class 1 component supports, Westinghouse uses the design criteria in Subsection NF, paragraph 3280. Stress limits for normal and upset conditions are those defined in Appendix XVII, paragraph 2460 of the Code, and Code Case 1644.

For emergency and faulted conditions, Westinghouse limits stresses to 0.9 S_y at temperature. For bolts supplied by Westinghouse, the ultimate strength is approximately ~~1500~~ 150 ksi and the yield strength is 130 ksi at temperature.

New Respon at 5/11/82 meetin

RAI 210.68

Describe to what extent high-strength NF bolts are used.

RESPONSE:

BOP

Approximately 5% of the NF supports use high strength bolts. Of these high strength bolted component supports, approximately 90% of the supports use friction type joints.

NSSS

Westinghouse only provides bolts for Class 1 component supports. Additional information concerning the properties of these bolts are discussed in the response to RAI 210.67.

New Response Revised at 5/14 mtg.

Revised

RAI 210.69

The Seabrook plant incorporates the Westinghouse Model F steam generator. We will require that the results of the analysis to determine the tube plugging criteria be presented to the staff when they become available.

RESPONSE:

Westinghouse will provide a report prior to commercial operation describing the tube plugging criteria which will be employed for Model F steam generators. Additionally, Westinghouse will review the current position in the Seabrook FSAR on Regulatory Guide 1.121 to confirm its applicability to the Seabrook plant.