SNUPPS
Standardized Nuclear Unit
Power Plant System

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September 5, 1981
SLNRC 81-090 File:0541
SUBJ: MEB Review

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos. STN 50-482, STN 50-483, and STN 50-486

Reference: NRC Summary of June 9-10, 1981 Meeting to Review Mechanical Engineering, dated July 6, 1981

Dear Mr. Denton:

On June 9-10, 1981, the SNUPPS applicants met with the NRC's Mechanical Engineering Branch. The reference reported the results of the meeting. The purpose of this letter is to provide the status of each meeting agenda item not previously resolved.

1. Agenda items 4 and 5 concerned pipe break analysis. SNUPPS FSAR Revisions 5 and 6 provided major updates to this analysis in FSAR Section 3.6. Although the pipe break analysis is not 100 percent complete, sufficient information is thought to be available for the NRC to draw satisfactory conclusions relative to the protection provided in the SNUPPS design against the hazards of postulated piping failures. This conclusion could be confirmed by the NRC in early 1982 when the analysis will be completed and the final FSAR updates to Section 3.6 are provided.

2. The enclosure to this letter provides FSAR changes that will be included in the next revision to the SNUPPS FSAR. These changes resolve meeting agenda items 16, 24, 25, 28, 34, 35, 37, and 38.

3. Agenda item 21 concerned the functional capability of piping. One clarification in the response to item 21 that was contained in the reference is required. In the third paragraph of the response it should be stated that the criteria that are used assure functional capability. The programs serve to check that the criteria are met.

4. Agenda item 29 concerned leak rate testing the isolation for low pressure systems connected to the reactor coolant pressure boundary (Event V). The SNUPPS report on this matter will be transmitted by a separate letter by September 9, 1981.

5. Agenda item 30 concerned the pump and valve test program. The program was submitted to the NRC by SLNRC 81-85, dated September 5, 1981.

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6. Agenda item 33 concerned an exception to Branch Technical Position MEB 3-1 that SNUPPS suggested at the June 9-10 meeting. Subsequent to the meeting, SNUPPS decided not to take this exception concerning splits in Class 1 branch lines. The FSAR accurately reflects SNUPPS position on the BTP, and therefore this matter should not be an issue.

6. NUREG-0737, item II.D.1 concerns testing of the pressurizer PORV. A change to the SNUPPS commitment concerning this item is included on revised page 18.2-24 of the enclosed FSAR changes.

7. NRC question MEB210.2 was transmitted to the SNUPPS applicants in a letter from the NRC's Mr. Youngblood dated August 25, 1981. A response to this question (concerning limiting the area of postulated pipe breaks) will be provided by separate letter by September 11, 1981.

Very truly yours,

R S Stright for

Nicholas A. Petrick

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Enclosure
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ENCLOSURE TO SLNRC 81-90
FSAR CHANGES

ME,3 #28

3.9(N).1.4.7 Stress Criteria for Class 1 Components and Component Supports

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 option outlined below:

a. The test 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.

The reactor vessel support pads are qualified using the test option. The reactor pressure vessel support pads are designed to restrain unidirectional horizontal motion, in addition to supporting the vessel. The design of the supports allows radial growth of the vessel but restrains the vessel from horizontal displacements, since tangential displacement of the vessel is prevented at each vessel nozzle.

To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a one-eighth linear scale model, were performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by a support shoe which was mounted to the test fixture.

The above modeling and application of load thus allows the maximum load capacity of the support pads to be accurately established. The test load, L_T, was then determined by multiplying the maximum collapse load by 64 (ratio of prototype area to model area) and including temperature effects in accordance with the rules of the ASME Code, Section III.

The loads on the reactor vessel support pads, as calculated in the system analysis for faulted conditions, are limited to the value of .80 L_T. The tests performed and the limits established for the test load method insure that the experimentally obtained value for L_T is accurate and that the support pad design is adequate for its intended function.

MEB 228

b. In the design of component supports, member compressive axial loads shall be limited to 0.67 times the critical buckling strength, per F-137c(c), of ASME III.

Loading combinations and allowable stresses for ASME Code, Section III, Class 1 components and supports are given in Tables 3.9(N)-2 and 3.9(N)-3.

The methods of load combination for each operating condition are as follows:

Design: Loads are combined by algebraic sum.

Normal, Upset: These loads are used in the fatigue evaluation in accordance with the methods prescribed in the ASME Code. Loadsets are defined for each transient, including the OBE, and are combined such that the maximum stress ranges are obtained without regard to the order in which the transients occur. (This is discussed in more detail in Section 3.9.1.4.3.)

Emergency: Loads are combined by algebraic sum.

Faulted: For primary equipment, primary equipment supports, and Class 1 branch lines, LOCA and SSE loads are combined using the square-root-of-the-sum-of-the-squares (SRSS) method on a load component basis (i.e. the LOCA F_x is combined with the SSE F_x by SRSS, the LOCA F_y is combined with the SSE F_y by SRSS, and likewise for F_z, M_x, M_y, and M_z). The sustained loads, such as weight effects, are combined with the SRSS result by algebraic sum.

For RCL piping, the deadweight moments were added to the LOCA moments prior to the SRSS combination of the LOCA and SSE loads.

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 systems during startup functional testing of the SNUPPS units. The purpose of these tests will be to confirm that the systems have 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 and steady state operation. If vibrations are observed which, from visual examination, appear to be excessive, either: 1) an

MEB#28
MEB#3

instrumented test program will be conducted and the system reanalyzed to demonstrate that the observed levels do not exceed one half the endurance limit at 10^6 cycles, defined in the ASME Code, 2) the cause of the vibration will be eliminated, or 3) the support system will be modified to reduce the vibrations. Particular attention will be provided at those locations where the vibrations are expected to be the largest for the particular transient being studied as per the criteria of the ASME Code referenced above.

It should be noted that the layout, size, etc., of the reactor coolant loop and surge line piping used on SNUPPS 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 loop and surge line piping are adequately designed and supported to minimize vibration. In addition, vibration levels of the reactor coolant pump, which is the only mechanical component that could cause vibration of the reactor coolant loop and surge line piping, are held to acceptable limits.

Thus, excessive vibration of the reactor coolant loop and surge line piping should not be present. However, as added assurance that excessive piping vibration is not present, the piping adjacent to the reactor coolant pump will be subjected to visual observation as discussed above.

3.9(N).2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The operability of Category I mechanical equipment must be demonstrated if the equipment is determined to be active, i.e., mechanical operation is relied on to perform a safety function. The operability of active Class 2 and 3 pumps, active Class 1, 2, or 3 valves, and their respective drives, operators, and vital auxiliary equipment is shown by satisfying the criteria given in Section 3.9(N).3.2. Other active mechanical equipment is shown operable by either testing, analysis, or a combination of testing and analysis. The operability programs implemented on the other active equipment are similar to the program described in Section 3.9(N).3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10(N) for electrical equipment are used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive seismic Category I equipment is shown to have structural integrity during all plant conditions in one of the following manners: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment or 2) by test showing that the equipment retains its structural integrity under the simulated test environment.

A list of seismic Category I equipment and the method of qualification used is provided in Table 3.2-1.

3.9(N).2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The vibration characteristics and behavior due to flow-induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed-form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independent of the

MEB 24

dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitations depends on many factors, such as type and location of component and flow conditions. The effects of these forcing functions have been studied from tests performed on models and prototype plants as well as component tests (Ref. 6, 7, 8, and 14).

The Indian Point No. 2 plant (Docket No. 50-247) has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant (Docket No. 50-344) instrumentation program and the Sequoyah No. 1 plant (Docket No. 50-327) instrumentation program provides prototype data applicable to SNUPPS (Ref. 6, 8, and 14).

The SNUPPS plants are similar to Indian Point No. 2; the only significant differences are the modifications resulting from the use of 17 x 17 fuel, replacement of the annular thermal shield with neutron shielding pads, and the change to the UHI-style inverted top hat support structure configuration. These differences are addressed below.

a. 17 x 17 fuel

The only structural change in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly is the guide tube. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow-induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation of internals vibration is expected from the vibration with the 15 x 15 fuel assemblies.

b. Neutron shielding pads lower internals

The primary cause of core barrel excitation is flow turbulence, generated in the downcomer annulus (Ref. 8). The vibration levels due to core barrel excitation for Trojan and SNUPPS, both having neutron shielding pads, are expected to be similar. The coolant inlet density of SNUPPS is slightly lower than Trojan, and the flow rate is slightly higher. Scale model tests show that the core barrel vibration varies as velocity is raised to a small power (Ref. 7). The difference in fluid density and flow rate

MEB 29

results in approximately 4 percent higher core barrel vibration for SNUPPS than for Trojan. However, scale model

test results (Ref. 7) and results from Trojan (Ref. 6) show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stresses and large safety margins were measured at Indian Point No. 2 (thermal shield configuration) lead to the conclusion that stresses less than or equal to those of Indian Point No. 2 will result on the SNUPPS internals.

c. UHI-style inverted top hat upper support configuration

The components of the upper internals are excited by turbulent forces due to axial and crossflows in the upper plenum and by pump-related excitations (Ref. 6 and 8). Sequoyah and SNUPPS have the same basic upper internals configuration; therefore, the general vibration behavior is not changed. The SNUPPS upper internals adequacy has been determined from data from instrumented plant tests at Sequoyah No. 1, scale model tests, and numerous operating plants. The results of testing at Sequoyah No. 1 (Ref. 14) showed that the components are excited by flow-induced and pump-related excitations. Analyses of the data indicate that the instrumented components have adequate factors of safety, the random flow-induced responses are adequately predicted by scale models, and that the margins are higher with the core in place than during hot functional testing.

In addition, the SNUPPS upper internals configuration was tested in scale model tests, using the same modeling techniques as for the scale model tests of the UHI configuration. The responses of the SNUPPS upper internals have been calculated using the Sequoyah No. 1 and scale model information. The results show adequate factors of safety for all components.

The original test and analysis of the four-loop configuration is augmented by References 6, 7, 8, and 14 to cover the effects of successive hardware modifications.

3.9(N).2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the SNUPPS reactor internals design configuration is well characterized, as was discussed in Section 3.9(N).2.3, it is not considered necessary to conduct instrumented tests of the SNUPPS hardware. The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory pre- and

MEB

post-hot functional examination for integrity. This examination will include in excess of 30 features illustrated in Figure 3.9(N)-3 with special emphasis on the following areas.

a. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

b. The lateral, vertical, and torsional restraints provided within the vessel.

c. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

MEB

d. Those other locations on the reactor internal components which are similar to those which were examined on the prototype Indian Point No. 2, and on Trojan and Sequoyah No. 1.

e. The inside of the vessel will be inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material are in evidence.

A particularly close inspection will be made on the following items or areas, using a 5X or 10X magnifying glass, where applicable.

a. Lower internals

1. Upper barrel to flange girth weld.

2. Upper barrel to lower barrel girth weld.

3. Upper core plate aligning pin. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.

4. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

5. Baffle assembly locking devices. Check for lockweld integrity.

6. Lower barrel to core support girth weld.

7. Neutron shielding pads screw locking devices and dowel pin lockwelds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity.

8. Radial support key welds.

9. Insert screw locking devices. Examine soundness of lockwelds.

10. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.

11. Secondary core support assembly weld integrity.

MEB 38

The stresses due to the SSE (vertical and horizontal components) were combined with the blowdown stresses by the SRSS method in order to obtain the largest principal stress and deflection.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to ensure control rod insertion. For the guide tubes deflected above the no-loss-of-function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9(N).2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

As stated in Section 3.9(N).2.3, it is not considered necessary to conduct instrumented tests of the SNUPPS reactor vessel internals. Adequacy of these internals are verified by use of the Sequoyah and Trojan results, supported by scale model tests. References 7 and 8 describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which utilize analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics so that the flow-induced vibratory behavior and response levels for SNUPPS are estimated. These estimates are then compared to values deduced from plant test data obtained from the Sequoyah and Trojan internals vibration measurement programs.

3.9(N).3 ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

The ASME Code Class components are constructed in accordance with the ASME Code, Section III.

A detailed discussion of ASME Code Class 1 components is provided in Section 3.9(N).1. For core support structures, design loading conditions are given in Section 4.2.2.3. Loading conditions are discussed in Section 4.2.2.4.

MEB #16

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, including thermal shock of the ECCS following a LOCA, 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.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For LOCA plus the SSE condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9(N)-12. The corresponding no-loss-of-function limits are included in Table 3.9(N)-12 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately 1/2 inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inches, which is insufficient to permit the tips of the RCCA to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident involving the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This structure limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated, assuming a complete and instantaneous failure of the primary core support, and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive stop is provided to ensure support.

5. Bogard, W. T. and Esselman, T. C., "Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants," WCAP-9279, March, 1978.

6. Bloyd, C. N., Ciaramitaro, W. and Singleton, N. R., "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8766 (Proprietary) and WCAP-8780, (Non-Proprietary), May, 1976.

7. Lee, H., "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," WCAP-8303-P-A (Proprietary) and WCAP-8317-A (Non-Proprietary), July, 1975.

8. Bloyd, C. N. and Singleton, N. R., "UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations, "WCAP-8516-P (Proprietary) and WCAP-8517 (Non-Proprietary), March, 1975.

9. Takeuchi, K., et al., "MULTIFLEX - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708-P-A, Volumes I and II (Proprietary) and WCAP-8709-A, Volumes I and II (Non-Proprietary), September, 1977.

10. Fabic, S., "Computer Program WHAM for Calculation of Pressure, Velocity and Force Transients in Liquid Filled Piping Networks," Kaiser Engineers Report No. 67-49-R, November, 1967.

11. Bohm, G. J. and LaFaille, J. P., "Reactor Internals Response Under a Blowdown Accident," First Intl. Conf. on Structural Mechanics in Reactor Technology, Berlin, September 20-24, 1971.

12. Cooper, F. W. Jr., "17 x 17 Driveline Components Tests Phase IB, II, III D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December, 1974.

13. Kraus, S., "Neutron Shielding Pads," WCAP-7870, June, 1972.

14. Altman, D. A., et al., "Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Tests on the Sequoyah 1 Power Plant," WCAP-9944 (Proprietary) and WCAP-9945 (Non-Proprietary), July 1981.

MEB

TABLE 3.9(N)-2

LOADING COMBINATIONS FOR ASME CLASS 1
COMPONENTS AND SUPPORTS (EXCLUDING PIPE SUPPORTS)

Condition Classification	Loading Combination
Design	Design pressure, design temperature, deadweight, Operating Basis Earthquake
Normal	Normal condition transients, deadweight
Upset	Upset condition transients, deadweight, Operating Basis Earthquake
Emergency	Emergency condition transients, deadweight
Faulted	Faulted condition transients, deadweight, Safe Shutdown Earthquake or Safe Shutdown Earthquake and Pipe Rupture Loads

TABLE 3.9(N)-3

ALLOWABLE STRESSES FOR ASME CODE, SECTION III, CLASS 1 COMPONENTS (a)(c)

Operating Condition Classification	Vessels/Tanks	Piping	Pumps	Valves	Component Supports(d)
Normal	NB-3222 (Level A)	NB-3653 (Level A)	NB-3222 (Level A)	NB-3525 (Level A)	NF-3222 NF-3231.1(a) (Level A)
Upset	NB-3223 (Level B)	NB-3654 (Level B)	NB-3223 (Level B)	NB-3525 (Level B)	NF-3223 NF-3231.1(a) (Level B)
Emergency	NB-3224 (Level C)	NB-3655 (Level C)	NB-3224 (Level C)	NB-3526 (Level C)	NF-3224 NF-3231.1(b) (Level C)
Faulted	NB-3225 (Level D)	NB-3656 (Level D)	NB-3225 (Level D)	(b)	NF-3225 NF-3231.1(c) (Level D)

(a) A test of the components may be performed in lieu of analysis.

(b) CLASS 1 VALVE FAULTED CONDITION CRITERIA

Active

a) Calculate P_m from para. NB3545.1 with Internal Pressure $P_s = 1.25P_s$

$P_m \leq 1.5S_m$

b) Calculate S_n from para. NB3545.2 with

$C_p = 1.5$

$P_s = 1.25P_s$

$Q_{t^2} = 0$

P_{ed} = 1.3X value of P_{ed} from equations of 3545.2(b) (1)

$S_n \leq 3S_m$

Inactive

a) Calculate P_m from para. NB3545.1 with Internal Pressure $P_s = 1.50\ P_s$

$P_m \leq 2.4S_m$ or $0.7\ S_u$

b) Calculate S_n from para. NB3545.2 with

$C_p = 1.5$

$P_s = 1.50\ P_s$

$Q_{t^2} = 0$

P_{ed} = 1.3X value of P_{ed} from equations of NB3545.2(b) (1)

$S_n \leq 3S_m$

P_s, P_e, P_m, P_b, Q_t, C_p, S_n & S_m, as defined by Section III of the ASME Code

(c) Limits identified refer to subsections of the ASME Code, Section III.

(d) Also see Appendix 3A, Regulatory Guides 1.124 and 1.130.

WEB #25

WEB #2B

TABLE 3.9(N)-4

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS [a] (EXCLUDING PIPE SUPPORTS)

Loading Combination[b,c]	Design/Service Level Requirements
1. Design pressure, design temperature, deadweight	Design
2. Normal condition pressure, normal condition metal temperature, deadweight, nozzle loads	Service Level A
3. Upset condition pressure, upset condition metal temperature, deadweight, nozzle loads, Operating Basis Earthquake	Service Level B
4. Emergency condition pressure, emergency condition metal temperature, deadweight, nozzle loads	Service Level C
5. Faulted condition pressure, faulted condition metal temperature, deadweight, nozzle loads, Safe Shutdown Earthquake	Service Level D

NOTES:

(a) The responses for each loading combination are combined using the absolute sum method. On a case-by-case basis, algebraic summation may be used when signs are known for final design evaluations.

(b) Temperature is used to determine allowable stress only.

(c) Nozzle loads, pressures, and temperatures are those associated with the respective plant operating conditions (i.e., normal, upset, emergency, and faulted), as noted for the component under consideration.

MEB #23

TABLE 3.9(N)-5

STRESS CRITERIA FOR SAFETY-RELATED ASME CLASS 2*
AND CLASS 3 VESSELS

Design/Service Level	Stress Limits**
Design and Service Level A	$\sigma_m \leq 1.0S$ (σ_m or σ_L) + $\sigma_b \leq 1.5S$
Service Level B	$\sigma_m \leq 1.1S$ (σ_m or σ_L) + $\sigma_b \leq 1.65S$
Service Level C	$\sigma_m \leq 1.5S$ (σ_m or σ_L) + $\sigma_b \leq 1.80S$
Service Level D	$\sigma_m \leq 2.0S$ (σ_m or σ_L) + $\sigma_b \leq 2.4S$

*Applies for tanks designed in accordance with the AS Code, Section III, NC-3300.

**Stress limits are taken from ASME III, Subsections NC and ND, or, for vessels procured prior to the incorporation of these limits into ASME III, from Code Case 1607.

MEB#28

TABLE 3.9(N)-6

STRESS CRITERIA FOR SAFETY-RELATED CLASS 2 VESSELS*

Design/Service Level	Stress Limits**
Design and Service Level A	$P_m \leq 1.0S_m$ $P_L \leq 1.5S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.5S_m$
Service Level B	$P_m \leq 1.1S_m$ $P_L \leq 1.65S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.65S_m$
Service Level C	$P_m \leq 1.2S_m$ $P_L \leq 1.8S_m$ $(P_m \text{ or } P_L) - P_b \leq 1.8S_m$
Service Level D	$P_m \leq 2.0S_m$ $P_L \leq 3.0S_m$ $(P_m \text{ or } P_L) + P_b \leq 3.0S_m$

*Applies for tanks designed in accordance with the ASME Code, Section III, NC-3200

**Stress limits are from ASME III, Subsection NC.

MEB# 23

TABLE 3.9(N)-7

STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3
INACTIVE PUMPS AND PUMP SUPPORTS

Design/Service Level	Stress Limits*
Design and Service Level A	$\sigma_m \leq 1.0S$ (σ_m or σ_L) + $\sigma_b \leq 1.5S$
Service Level B	$\sigma_m \leq 1.1S$ (σ_m or σ_L) + $\sigma_b \leq 1.65S$
Service Level C	$\sigma_m \leq 1.5S$ (σ_m or σ_L) + $\sigma_b \leq 1.80S$
Service Level D	$\sigma_m \leq 2.0S$ (σ_m or σ_L) + $\sigma_b \leq 2.4S$

*Stress limits are taken from ASME III, Subsections NC and ND, or, for pumps procured prior to the incorporation of these limits into ASME III, from Code Case 1636.

TABLE 3.9(N)-8

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

Design/Service Level	Design Criteria*	
Design, Service Level A and Service Level B	$\sigma_m \leq 1.0S$	$\sigma_m + \sigma_b \leq 1.5S$
Service Level C	$\sigma_m \leq 1.2S$	$\sigma_m + \sigma_b \leq 1.65S$
Service Level D	$\sigma_m \leq 1.2S$	$\sigma_m + \sigma_b \leq 1.8S$

*The stress limits specified for active pumps are more restrictive than the ASME III limits to provide assurance that operability will not be impaired for any operating condition.

MEB
#28

TABLE 3.9(N)-9

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES

Design/Service Level	Stress Limits (a,b,c,d, and f)	P_{max} (e)
Design and Service Level A	Valve bodies shall conform to ASME Code, Section III	1.0
Service Level B	$\sigma_m \leq 1.1S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$	1.0
Service Level C	$\sigma_m \leq 1.5S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80S$	1.2
Service Level D	$\sigma_m \leq 2.0S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4S$	1.5

NOTES:

(a) Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied: 1) the section modulus and area of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and 2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, or connected piping material. If the valve body material allowable stress is less than that of the connected piping, the required acceptance criteria ratio shall be 110 percent multiplied by the ratio of the pipe allowable stress to the valve allowable stress. If unable to comply with this requirement, an analysis in accordance with the design procedure for Class 1 valves is an acceptable alternate method.

(b) Casting quality factor of 1.0 shall be used.

(c) These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

MEB #23

TABLE 3.9(N)-9 (Sheet 2)

(d) Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

(e) The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure. If these pressure limits are met, the stress limits in this table are considered to be satisfied.

(f) Stress limits are taken from ASME III, Subsections NC and ND, or, for valves procured prior to the incorporation of these limits into ASME III, from Code Case 1635.

ME3 # 37

TABLE 3.9(B)-3

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MEB #37

TABLE 3.9(B)-4

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MEB#34

REGULATORY GUIDE 1.123 REVISION 1 DATED 7/77

Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A in each Site Addendum.

REGULATORY GUIDE 1.124 REVISION 1 DATED 1/78

Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports

DISCUSSION:

According to the NRC implementation guidance for this regulatory guide, it is not applicable to the SNUPPS units. However, the following discussion is provided for information purposes.

For ASME Section III components not supplied with the NSSS, the recommendations of this regulatory guide are met as discussed in Table 3.9(B)-14.

The Westinghouse position with respect to this regulatory guide is as follows.

a. The Regulatory Guide states in Paragraph B.1(b): "Allowable service limits for bolted connections are derived from tensile and shear stress limits and their nonlinear interaction; they also change with the size of the bolt. For this reason, the increases permitted by NF-3231.1, XVII-2110(a), and F-1370(a) of Section III are not directly applicable to allowable shear stresses and allowable stresses for bolts and bolted connections," and in Paragraph C.4: "This increase of level A or B service limits does not apply to limits for bolted connections."

   As noted above, the increase in bolt allowable stress under emergency and faulted conditions is not permitted. Westinghouse believes that the present ASME Code rules are adequate for bolted connections. This position is based on the following:

   It is recognized after extensive experimental work by several researchers that the interaction curve between the shear and tension stress in bolts is more closely represented by an ellipse and not a line. This has been clearly recognized by the ASME. Code Case 1644-6 specifies stress limits for bolts and represents this tension/shear relationship as a nonlinear interaction equation (incorporated into ASME III, Appendix XVII via the Winter 77 Addenda) and has a built-in

MEB#34

safety factor that ranges between 2 and 3 (depending on whether the bolt load is predominantly tension or shear) based on the actual strength of the bolt as determined by test (Ref: "Guide to Design Criteria of Bolted and Riveted Joints," Fisher and Struik, copyright 1974, John Wiley and Sons, Page 54).

Study of three interaction curves of allowable tension and shear stress based on the ASME Code (emergency condition allowables per XVII-2110 and faulted condition allowables per F-1370) and the ultimate tensile and shear strength of bolts (obtained from experimental work published by E. Chesson, Jr., N. L. Faustino, and W. H. Munse, "High Strength Bolts Subjected To Tension and Shear," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, October 1965, Pages 155-180) indicates that there is adequate safety margin between the emergency and faulted condition allowables and failure of the bolts.

During their tests to determine the strength and behavior characteristics of single high strength bolts subjected to various combinations of tension and shear (T-S), Chesson, et. al. used a total of 115 bolts to ASTM Specificaions A 325-61T and A 354-Grade BC. The A 325-61T, which is a medium carbon steel, had a yield point of 77,000 psi to 88,000 psi and ultimate strength of 105,000 psi to 120,000 psi, depending upon the bolt diameter. The A 354-Grade BC, which is a heat treated carbon steel, had a yield point of 99,000 psig to 109,000 psi and ultimate strength from 115,000 psi to 125,000 psi, again depending upon the bolt diameter.

Figure 3A-2 shows the interaction curves for T-S loads on SA-325 bolts. Curve (1) represents the interaction relation (ellipse) permitted by Code Case 1644 (ASME III, Appendix XVII Winter 77 Addenda) for service levels A, B, and design condition. Curve (2) represents the interaction curve which considers the Code Case 1644 allowables and the increase permitted by XVII-2110(a) for service level C. Curve (3) represents the interaction curve which considers the Code Case 1644 allowables and the increase permitted by F-1370(a) for service level D. Curve (3) is the upper limit of the allowable stresses.

The design stress limits represented by Curves 1, 2, and 3 for A 325 bolts are then compared against the ultimate strength of the bolts represented by Curve 4, which is based on Chesson's test results. The area between Curve 3 and Curve 4 is the safety margin between the maximum bolt stress under service level D and minimum ultimate strength of the bolt.

MEB#39

Factor of safety against failure for A 325 bolts for various T-S ratios is shown in Figure 3A-3. The safety factor varies between a minimum of 1.36 and a maximum of 2.29, depending upon the value of T-S ratio. This is based upon the ultimate strength of the bolts from Chesson's test and the allowables obtained from Code Case 1644 and the increase permitted by F-1370(a) for service level D. Figure 3A-3 demonstrates that there exists an adequate factor of safety for the complete range of T-S loadings.

From this study it is observed that:

(1) For the emergency condition, the safety factor (ratio of ultimate strength to allowable stress) varies between a minimum of 1.63 and a maximum of 2.73, depending upon the actual tensile stress/shear stress (T/S) ratio on the bolt.

(2) For the faulted condition, the safety factor varies between a minimum of 1.36 to a maximum of 2.29, again depending upon actual T/S ratio on the bolt.

It is thus reasonable to allow an increase in these limits for the emergency and faulted conditions.

The Westinghouse design of component supports restricts the use of bolting material to the following applications:

(a) Westinghouse design uses bolting predominantly in tension. Oversized holes are generally provided, and a mechanism other than the bolts is provided to take any shear loads. Shear or shear and tension interaction occur only in isolated locations:

(b) Westinghouse bolts are limited to the following materials: A490, SA-354, SA-325, SA-540.

(c) The diameters used range between 1/2 inches and 3 inches.

For the emergency condition, Westinghouse will use allowable bolt stresses specified in Code Case 1644, as increased according to the provisions of XVII-2110(a).

For the faulted condition, tensile loads in the bolts shall be limited to 0.7 Su, but not to exceed in any case 0.9 Sy. The allowables are taken at temperature. In those few cases where bolts are used in shear or tension and shear, ASME Code Appendix XVII - 2460 Requirements will apply with an increase factor that

MEB 34

is defined in Regulatory Guide 1.124 or in Appendix F-1370, whichever is more restrictive. This provides an adequate margin of safety for the Westinghouse design.

b. In Paragraphs B.5 and C.8 of the Regulatory Guide, Westinghouse takes exception to the requirement that systems whose safety-related function occurs during emergency or faulted plant conditions, must meet level B limits. The reduction of allowable stresses to no greater than level B limits (which in reality are design limits since design, level A, and level B limits are the same for linear supports) for support structures in those systems with safety-related functions occurring during emergency or faulted plant conditions is overly conservative. The primary concern is that the system remains capable of performing its safety function. For active components, this is accomplished through the operability program, as discussed in Section 3.9N.3.2.

c. Paragraph C.6(a) of the Regulatory Guide appears confusing as to what stress limits may be increased for the emergency condition. Westinghouse will interpret this paragraph as follows: "The stress limits of XVII-2000 of Section III and Regulatory Position 3, increased according to the provisions of XVII-2110(a) of Section III and Regulatory Position 4, should not be exceeded for component supports designed by the linear elastic analysis method."

d. The method described in Paragraph C.7(b) of the Regulatory Guide is overly conservative and inconsistent with the stress limits presented in Appendix F. Westinghouse will use the provisions of F-1370(d) to determine service level D allowable loads for supports designed by the load rating method.

REGULATORY GUIDE 1.125 REVISION 1 DATED 10/78

Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of each Site Addendum.

REGULATORY GUIDE 1.126 REVISION 1 DATED 3/78

An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

MEB#39

DISCUSSION:

The fuel densification model used for the SNUPPS units is presented in "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP-8219-A (Non-Proprietary), March, 1975, which have been approved by the NRC.

MFB #34

REGULATORY GUIDE 1.127 REVISION 1 DATED 3/78

Inspection of Water-Control Structures Associated with Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of each Site Addendum.

REGULATORY GUIDE 1.128 REVISION 1 DATED 10/78

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 8.3.2.2.1.

REGULATORY GUIDE 1.129 REVISION 1 DATED 2/78

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 8.3.2.2.1.

REGULATORY GUIDE 1.130 REVISION 1 DATED 10/78

Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports

DISCUSSION:

According to the NRC implementation guidance for this regulatory guide, it is not applicable to the SNUPPS units. However, the following discussion is provided for information purposes.

For ASME Section III components not furnished with the NSSS, the Class 1 supports are of the linear type and not the plate and shell type. Therefore, this regulatory guide does not apply.

The Westinghouse position with respect to this regulatory guide is as follows.

a. Paragraph B.1 states that increases are not allowed for bolted connections for emergency and faulted conditions. The Westinghouse position is that it is reasonable to allow an increase in the limits for

MEM#39

bolted connections for these conditions. Further justification concerning this position can be found in Item 1 of the discussion on Regulatory Guide 1.124.

b. The method described in Paragraph C.7(b) of the Regulatory Guide is overly conservative and inconsistent with the stress limits presented in Appendix F. Westinghouse will use the provisions of F-1370(d) to determine service level D allowable loads for supports designed by the load rating method.

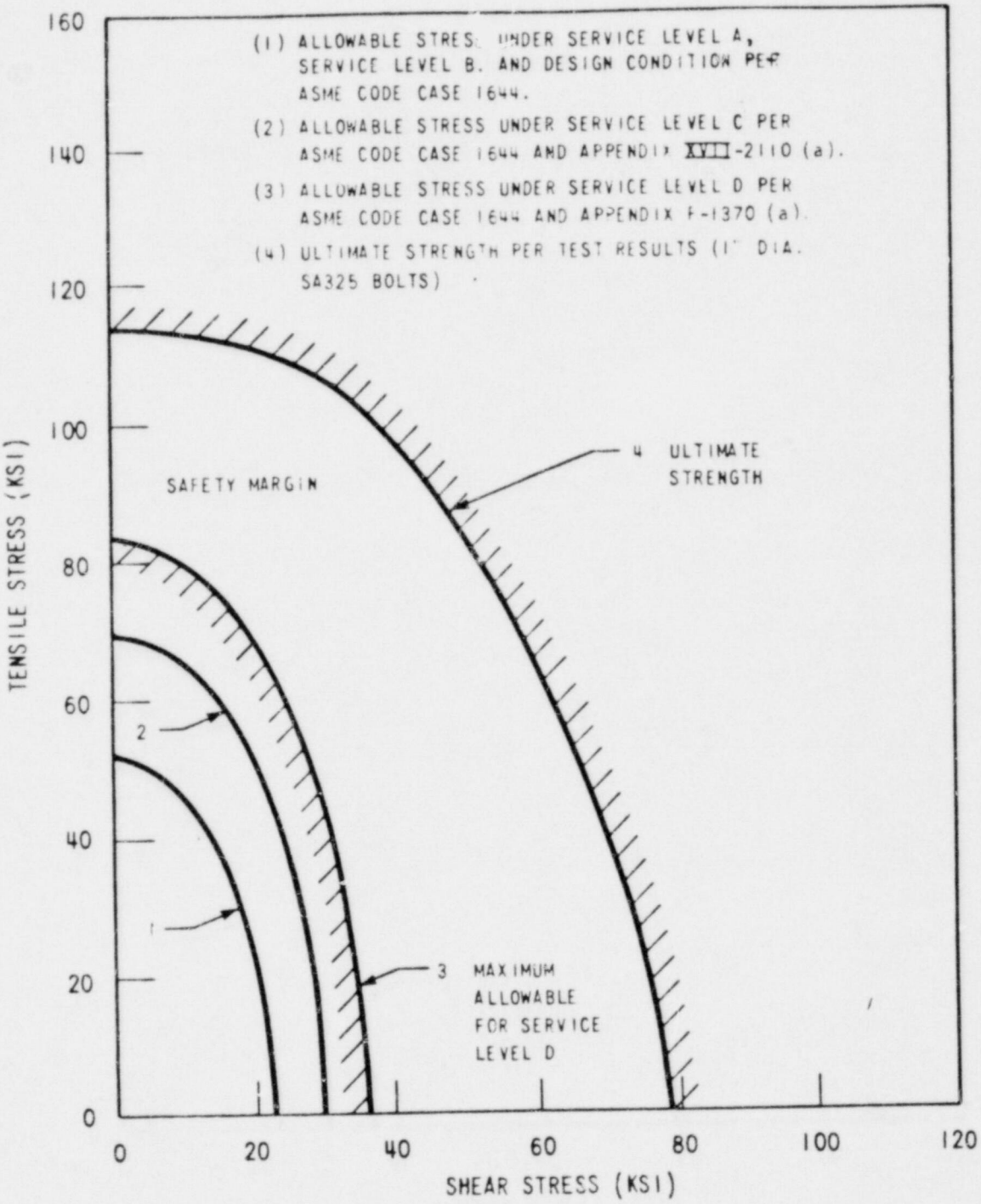

Figure ~~18-1~~ 3A-2

Comparison of Tensile Stress for Bolts

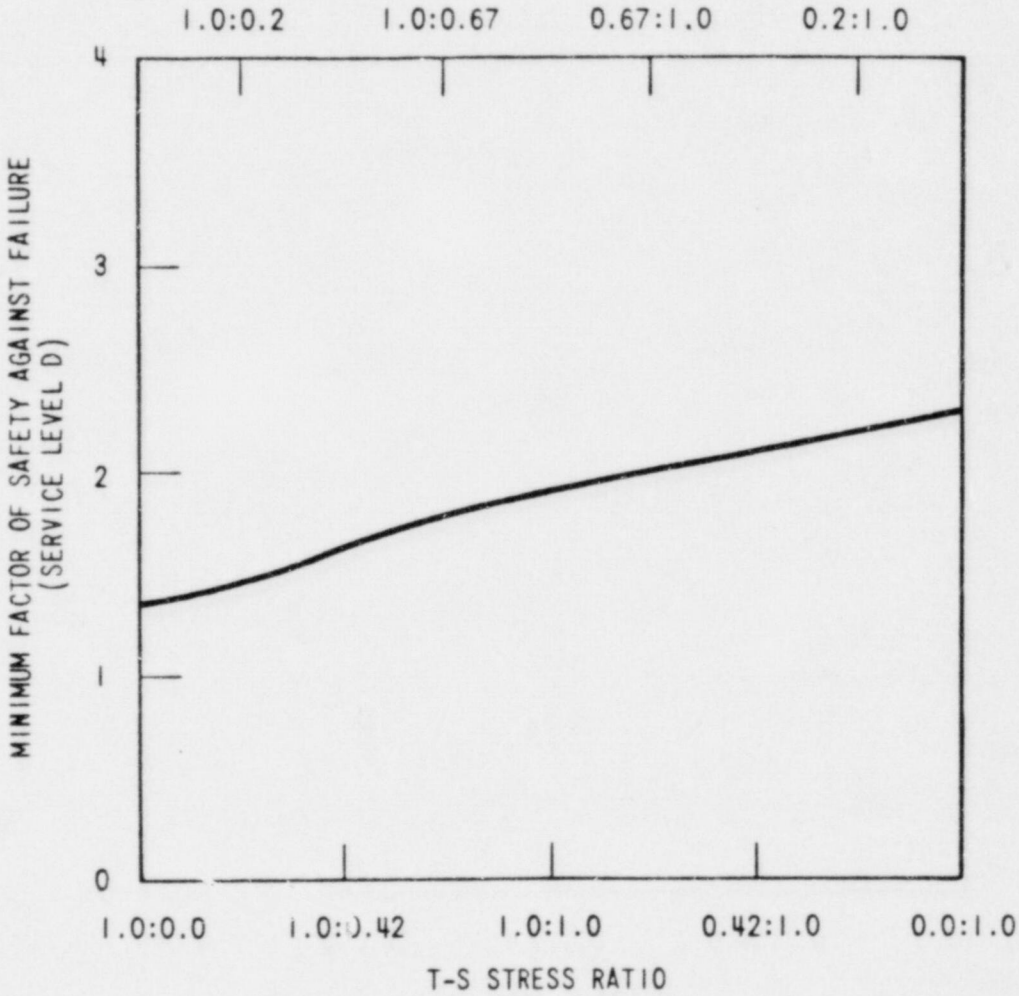

Figure ~~1.8.2~~ 3d 3

Factor of Safety Against Failure Under Service Level D as a Function of T-S Ratio

Block valve testing

initiated with a water-solid inlet (loop seal) at the PORVs and a steam bubble maintained in the pressurizer.

Regarding verification of the block valve functionability, the topic is under discussion between PWR utilities and the NRC staff. ~~SNUPPS plans to abide by the final conclusions reached by the PWR and the NRC.~~

18.2.5.5 Conclusion

The SNUPPS plan to demonstrate the operability of the PORVs and safety valves in the SNUPPS plants satisfies the guidance of item II.D.1 in NUREG-0737.

SNUPPS will provide evidence to show the qualification of the block valves by July 1, 1982. SNUPPS plans are to demonstrate qualification by use of the generic report to be submitted by PWR utilities in cooperation with EPRI.