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SUBJECT: Forwards marked-up response to draft SER concerns & branch meeting issues,as result of discussions w/NRC reviewers.

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Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
SUBMITTAL OF SER OPEN ISSUES

Enclosed is our submittal responding to draft SER concerns and branch meeting issues, as a result of informed discussions with Supply System Licensing personnel in Bethesda and NRC reviewers.

It is the intent of the Supply System to furnish this information as an assist to the NRC Project Manager to close-out issues on the pending WNP-2 SER.

Very truly yours,

GD Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

RMN/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

B001
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SUBMITTAL OF SER OPEN ISSUES

Reference	Issue	Resolution
	Question 421.043 - Q-List	FSAR Change
SEB-21	DG oil storage tank and connecting piping relative displacement.	FSAR Change
Page 3.7-12	Delete "...of closely spaced modes..."	FSAR Change
SEB-34	Submit calculations for under-ground piping	Will be sent from B&R on 02/08/82
6.7	Add sentence "Leakage in the MSIV System is designed to conform with the leakage rate specified in the Standard Technical Specifications."	FSAR Change
	Question 010.055 - Added Light Loads over the reactor vessel table. Clarify Items 17 and 22 of Light Loads over the Spent Fuel Pool.	FSAR Change
PSB	Question 040.086 - Added installation date of air dryer assembly	FSAR Change
PSB	Question 040.046 - NRC requested additional emergency lighting	Response Clarified
PSB	Question 040.017 - NRC requested additional information	Response Clarified
PSB	Question 040.032 - Change response to read inspection of valves will be observed every 31 days	Response Clarified - Change to Technical Specification already made.

9.	Instrumentation for detection of inadequate core cooling	II.F.2
10.	HPCI & RCIC initiation levels	II.K.3.13
11.	Isolation of HPCI & RCIC	II.K.3.15
12.	Challenges to and failure of relief valves	II.K.3.16
13.	ADS actuation	II.K.3.18
14.	Restart of core spray and LPCI	II.K.3.21
15.	RCIC suction	II.K.3.22
16.	Space cooling for HPCI & RCIC	II.K.3.24
17.	Power on pump seals	II.K.3.25
18.	Common reference level	II.K.3.27
19.	ADS valve, accumulators, and associated equipment and instrumentation	II.K.3.28
20.	Emergency plans	III.A.1.1/ III.A.2
21.	Emergency support facilities	III.A.1.2
22.	In-plant I_2 radiation monitoring	III.D.3.3
23.	Control room habitability	III.D.3.4

Response:

~~Table 3.2-1 will be modified prior to fuel load to add and~~ ~~identify the~~ quality class of each generic item listed in the question and not already included in the table. Work performed during the operating phase, including modification, maintenance, calibration, and testing, ~~will be~~ performed under ~~the applicable requirements of the Operational Quality Assurance Program.~~

Included

a. The following responses correspond with the same numbered items as in Question 421.043:

1. The biological shielding is part of the structures of the Reactor Building, Containment and Control/Radwaste Buildings. (See FSAR Section 3.8.2, 3.8.3 and 3.8.4.1.1.). As indicated in Table 3.2-1, Items 46 and 47, applicable parts of these sturctures are QC-I and thus all modifications to the biological shielding will be performed under the appropriate QA measures. Further addition to Table 3.2-1 is not required.

2. All missile barriers with the exception of the RPS MG set barrier are part of structures. Use of structural walls for barriers is covered in FSAR Section 3.5. The RPS MG set missile barrier is addressed in a forthcoming revision to FSAR Section 3.5 and is safety-related and controlled by the QA Program. Since where missile barriers are required, they are part of Seismic Category I structures they are QC-I as addressed in Item 46 and 47 of Table 3.2-1. Thus all modifications to missile barriers will be performed with the appropriate QA measures.

3. The spent fuel pool is safety-related and is part of the Reactor Building structure (FSAR 3.8.4.1.1.6) and is covered in Table 3.2-1 under Item 47.1. The spent fuel pool liner is safety-related and is addressed in Item 34.1 of Table 3.2-1. Further addition to Table 3.2-1 is not required.

4. Equipment and drain floor piping and containment isolation valves are addressed in Item 19.3 (refer to color coded Figures 3.2-9, 3.2-10 and Figure 11.2-2).* Further additions to Table 3.2-1 are not required.

5. Quenchers and quencher supports are safety-related and under QA Program requirements and listed in Table 3.2-1 under Item 2.4 (refer to color coded Figure 3.2-2). (Supports always meet the same or higher QA requirements as the item supported.)

6. Downcomers and braces are safety-related and under QA Program requirements and listed in Table 3.2-1 under Item 2.4 (refer to color coded Figure 3.2-2).

7. The containment spray system is part of the Residual Heat Removal (RHR) system, is safety-related and under QA Program requirements and listed under Items 10.4 and 10.8 (refer to color coded Figure 3.2-6) in Table 3.2-1.

8. Condensate and feedwater piping from PRV to the outermost isolation valves and the containment isolation valves are safety-related and are under QA Program requirements and are listed in Table 3.2-1 under Items 2.5 and 2.11 (refer to color coded Figure 3.2-2).

9. Primary containment access hatches/locks/doors are safety-related, are attached to the containment vessel and subject to QA Program requirements. These items are covered by Item 46 "Containment Vessel" in Table 3.2-1. Figure 3.2-1 shows that everything pertaining to the containment boundary would be code Group B, QC-I. Accordingly, primary containment penetration assemblies are safety-related and are under QA Program requirements. They are specifically addressed in FSAR Section 3.8.6 and 3.8.2.2.4.

*Color coded figures are FSAR figures.

Primary Containment vacuum relief valves are safety-related and under QA Program Management. They are part of the Primary Containment cooling and purging system and covered under Item 28 of Table 3.2-1, see color code Figure 3.2-15.

10. Engineering safety features actuation systems are safety-related and are under QA Program Management. These instrument and control systems are addressed in FSAR Sections 7.3 and 7.4 and are covered in Table 3.2-1 under each applicable system, i.e., for HPCS Item 12.10 covers the electrical components of the Engineering safety features actuation system for the HPCS system.

11. Combustible gas control system hydrogen recombiners are safety-related and under QA program control and are addressed in Item 30 of Table 3.2-1, refer to color coded Figure 3.2-17.

12. Safety-related instrument and control systems are identified in Chapter 7, Table 7.1-1, of the FSAR and are under QA program control. A footnote to this effect will be added to Table 3.2-1.

13. All of the items in Section 13 a) through l) are safety-related and controlled by the QA Program with the following clarifications.

    a) Diesel generator packages, including auxiliaries, are safety-related to the extent as defined in FSAR Table 3.2-1, Item 38.

    b) Valve operators are considered with the valves where they are installed and are addressed in Table 3.2-1 under the system the valve is installed in.

    c) Conduit and cable trays and their supports for Class IE cables and those whose failure may damage other safety-related items are safety-related and controlled by the QA Program.

    d) Instrumentation, control, power cables, transfers, inverters, etc., are considered with the system for which they are installed. If the system is a safety-related system it is controlled by the QA Program.

    e) Fire-rated penetration seals for cable systems will be under the control of the Supply System Operational QA Program.

14. All of the items in Section 14 a) through d) are safety-related and controlled by the QA Program with the following clarifications.

    a) Conduit and cable trays and their supports for Class IE cables and those whose failure may damage other safety-related items are safety-related and controlled by the QA Program.

    b) Battery racks are considered with the batteries.

    c) Protective relays and control panels are considered with the equipment panel they service. (Item 13d above is applicable.)

15. The normal operation fixed area and airborne montioring systems are discussed in FSAR Subsection 12.3.4. These systems are not safety-related and, therefore, are not controlled by the QA Program.

The post-accident high range radiation monitoring system for the drywell and containment is safety-related and the components are controlled by the QA Program.

Portable radiation monitoring is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of these monitors as well as calibration of all radiation monitors is provided for by the appropriate WNP-2 Administrative Procedures. These procedures are subject to the pertinent requirements of the Supply System Operational QA Program.

16. The normal operation and post-accident process and effluent radioactivity monitoring systems are discussed in FSAR Section 7.5 and 11.5.

    The only radioactivity monitoring components that are controlled by the QA Program are the radiation monitors for the main steam line, reactor building ventilation monitor, and the containment atmosphere radiation monitor. These monitors are covered in Items 9 and 48 of Table 3.2-1.

    Portable radioactivity monitoring is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of these monitors as well as calibration of all radioactivity monitors is provided by the appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

17. The normal sampling systems are discussed in FSAR Subsections 9.3.2 and 12.3.4. These systems are not safety-related and are not, therefore, controlled by the QA Program. No revision to Table 3.2-1 is required.

18. Radioactive contamination measurement and analysis is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

19. Personnel monitoring internal and external is not a "structure, system or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

20. As required by the Supply System Operational Quality Assurance Program, WNP-2 has in-place measures to assure that measuring and testing equipment used in activities affecting quality are stored, controlled, calibrated and adjusted to maintain accuracy within specified limits.

21. Decontamination is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

22. Respiratory protection including testing is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

23. Contamination control is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

24. Radiation shielding at WNP-2 may be classified as 1) shielding required to limit off-site radiation doses to allowable limits, and 2) shielding required to limit in-plant doses for personnel access to various plant areas.

Radiation shielding to limit off-site doses is considered safety-related and is provided by the containment and auxiliary buildings. These structures are fully designed as safety-related structures and are capable of withstanding all postulated natural phenomena and dynamic events.

Radiation shielding for personnel access to various plant areas is not considered safety-related. Radiation shielding for this purpose is provided in containment, turbine building, and radwaste/control building. Reinforced concrete walls are used to provide for necessary shielding. The in-plant radiation shielding walls in the radwaste/control building and containment are considered safety-related only to the extent that they must maintain structural integrity, i.e., the radiation shielding capability is not safety-related.

The radiation shielding walls in the turbine building have no safety-related function.

The quality assurance requirements for shielding are commensurate with QA requirements for the structures in which it is located. The QA requirements for the Containment, Reactor and Radwaste/Control Buildings are given in FSAR Table 3.2-1, Sections 46 and 47, Item 1 and 3.

25. Meteorological data collection equipment is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate WNP-2 Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

26. This item is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate Administrative Procedures. These procedures are subject to pertinent requirements of the Supply System Operational QA Program.

27. As required by the Supply System Operation Quality Assurance Program, WNP-2 has in-place measures to assure that measuring and testing equipment used in activities affecting quality are stored, controlled, calibrated and adjusted to maintain accuracy within specified limits.

28. WNP-2 has no safety-related masonry walls.

29. Class IE electrical duct banks are safety-related and under the control of the QA Program.

30. WNP-2 essential service water pipe line is the standby service water system, and safety-related piping, including buried piping, is under QA Program requirements. This item is covered in Table 3.2-1, Section 25.1.

b. The following responses correspond with the same numbered items as in Question 421.043:

1. WNP-2 FSAR update (Amendment 20) upgrades the spent fuel pool cooling function of the Fuel Pool Cooling and Cleanup system to Quality Class I and Table 3.2-1 will be updated accordingly. The spent fuel pool cleanup function of the Fuel Pool Cooling and Cleanup system is not safety-related and therefore is not covered by the QA Program.

2. Item 1.6 of FSAR Table 3.2-1 includes these non-safety class internal structures such as feedwater spargers, steam dryers, shroud bead and steam separator assembly, incore guide tubes and stabilizers, and surveillance sample holders. These structures do not perform a safety function and are not required to prevent or mitigate the consequences of accidents. A failure of the feedwater sparger will not prevent transmission of cooling water to the core affecting the safety of the reactor system. Although these structures are not safety-related, they are so designed that they will not adversely affect the safety function of the safety-related structures. These non-safety structures are installed under QA Program requirements, and maintenance performed in the reactor vessel on these components is performed utilizing quality-affecting procedures which are under the control of the operational QA Program.

c. The following responses correspond with the same numbered items as in Question 421.043:

1. The plant-safety-parameter display console is not safety-related. Justification is contained in NUREG-0696 Paragraph 2.5 and 4.2 (Table and Footnotes). Emergency Facilities (1) are not required for safe shutdown or immediate or long-term operation following a LOCA, and (2) will not cause the release of radioactivity in excess of 10CFR100 limits or increase severity of a DBA if they should fail. Therefore, Emergency Support Facilities will-not be added to Table 3.2-1.

2. Vents are not required on BWR's to ensure post-accident natural circulation capability (see FSAR Appendix B Section II.B.1), but are provided for other uses. The vents are located in existing safety-related piping systems. No modifications in design were required to meet the requirements of this item. No change is required to FSAR Table 3.2-1 in that the vents are already shown on the various color coded figures.

3. The plant shielding item requires a <u>review</u> of the accessibility of various station areas under post-accident conditions. This review is not a "structure, system, or component" and thus is not appropriate for Table 3.2-1.

4. The post-accident sampling system is currently in the design stage. Revisions to the color coded figures will show the appropriate quality class when the design is finalized.

5. As stated in FSAR Appendix B Section II.D.3, a safety/relief valve position monitoring system is being added to WNP-2 to indicate the open/closed condition of each safety/relief valve. The system will meet the same quality requirements as stated for Section 2, Item 14 of FSAR Table 3.2-1.

6. Dedicated hydrogen penetrations are safety-related and included in Item 30 of Table 3.2-1 (refer to color coded Figure 3.2-17).

7. The containment isolation valves and their associated circuits are safety-related and controlled by the QA Program. The isolation valves are listed in FSAR Table 3.2-1 under each individual applicable system. See also the response to Item a.10 above.

8. Modifications to the Accident Monitoring System are addressed in FSAR Appendix B Section II.F.1.1. Parts of the design modifications required by this item are safety-related and will result in modifications to Chapter 7 of the FSAR. The safety-related portions of these design modifications will be under the QA Program.

9. WNP-2 is performing a study in response to this item, the safety relatedness of any additional instrumentation systems that may result from this study will be determined when the study is complete. See FSAR Appendix B Section II.F.2.

10. As stated in FSAR Appendix B Section II.K.3.13, there is no change planned in HPCS and RCIC initiation levels. As stated in this section; however, modifications will be made for auto-reset of RCIC, this addition will meet the same requirements of FSAR Table 3.2-1, Item 8.

11. As stated in FSAR Appendix B Section II.K.3.15, a time delay to the RCIC break detection circuitry will be added. This addition will meet the same quality requirement as given in FSAR Table 3.2-1 Section 13 Item 8.

12. FSAR Appendix B Section II.K.3.16 indicates that further modification to the WNP-2 design would not significantly reduce the frequency of safety/relief valve events. Therefore, no changes to FSAR Table 3.2-1 are required.

13. As stated in FSAR Appendix B Section II.K.3.18, no changes to the ADS is required, therefore, no change to FSAR Table 3.2-1 is required.

14. As stated in FSAR Appendix B Section II.K.3.21, modification to provide automatic restart for core spray and LPCI is not required. Therefore, change to FSAR Table 3.2-1 is not required.

15. The automatic switchover of the RCIC suction from the condensate storage tank to the suppression pool is considered safety-related and is subject to the pertinent QA requirements for Class IE electrical systems. Appropriate changes to Table 3.2-1 and the associated figures will be made when the design is finalized.

16. FSAR Appendix B Section II.K.3.24 describes the emergency space cooling system to the equipment rooms containing the HPCS and RCIC pumps. No design changes were necessary to meet this item, appropriate components of this system are under QA Program control and are listed in Table 3.2-1.

17. As stated in FSAR Appendix B Section II.K.3.25, no change in the WNP-2 design is required. Therefore, no addition to FSAR Table 3.2-1, is required.

18. As stated in FSAR Appendix B Section II.K.3.27, no change, other than recalibration, in the WNP-2 design is required. Therefore, no addition to FSAR Table 3.2-1 is required.

19. All equipment associated with the ADS System is safety-related and controlled by the QA Program. Major components are listed in FSAR Table 3.2-1 Section 37.

20. Emergency plans are not a "structure, system or component" requiring entry in Table 3.2-1. Emergency plan procedures are subject to audit by Supply System QA.

21. Equipment and other items associated with the Emergency Support Facilities are not safety-related. Justification is contained in NUREG-0696 Paragraph 2.5 and 4.2 (Table and Footnotes). Emergency Facilities (1) are not required for safe shutdown or immediate or long-term operation following a LOCA, and (2) will not cause the release of radioactivity in excess of 10CFR100 limits or increase severity of DBA if they should fail. Therefore, Emergency Support Facilities will not be added to Table 3.2-1.

22. Inplant I_2 radiation monitoring would be performed under post-accident procedures in accordance with the Supply System Emergency Plan. (See FSAR Appendix B Section III.D.3.3.). Equipment for this monitoring under post-accident conditions will be controlled by procedures subject to the pertinent requirements of the Supply System Operational QA Program.

23. The control room HVAC system is safety-related and controlled by the QA Program. This system is addressed in FSAR Table 3.2-1 Section 31.

ISSUE #21

Provide an assessment of the effect of the relative displacement between the DG oil storage tank and connecting piping and the additional stresses which could be produced by the seismic wave passage.

RESPONSE

~~The assessment of the small diameter piping associated with the DG oil storage tank is currently in progress, and the information requested will be furnished in January, 1982.~~

An assessment has been made of the effect of the relative seismic displacement between the diesel generator oil storage tank and the buried connecting piping, and the additional stresses which could be produced by the seismic wave passage. The only piping involved is small diameter piping connecting the diesel oil storage tank to the day tanks inside the diesel generator building.

Axial and bending stresses due to seismic soil strain, differential movement of pipe and structure, temperature and internal pipe pressure have been calculated for both straight pipe and pipe bends.

This piping is Quality Class I, Seismic Category I and ASME Section III, Class 3. The analyses demonstrate the adequacy against the applicable design conditions.

Stresses do not exceed ASME Section III allowables and design margins are calculated. The design margins (allowable divided by actual) vary from a minimum of 1.51 to a maximum of 2.98, depending on the design condition.

The methodology employed in the assessment is the same as documented in FSAR Section 3.7.3.12 revised in Amendment 8 of the FSAR.

Q. 130.020

In Section 3.7.2.1.8.3.1 of the FSAR, you state that the stresses obtained for each natural mode are superimposed for all modal displacements of the structure using the SRSS methodology. Indicate whether there are any closely-spaced modes as defined by Equation 3.7.2.1-13. If so, indicate how the responses of the closely-spaced modes were combined with other modal responses in your calculations.

Response:

~~The NSSS seismic design of WNP-2 was established prior to the issuance of Regulatory Guide 1.92, therefore, the consideration of closely-spaced modes in the response spectrum method of seismic analysis as described in this regulatory guide was not a requirement for the issuance of the WNP-2 construction permit. Hence, for the NSSS equipment where the response spectrum method of seismic analysis was used the closely-spaced modal responses were combined by the SRSS method. However, for all current seismic analyses of the main steam and reactor recirculation ASME Safety Class 1 piping the double sum method approved by the NRC on GESSAR 251 docket is used to combine the closely-spaced modal responses. Where the equivalent static load method and the time-history method of seismic analysis are used to calculate maximum responses it precludes the need to consider closely-spaced modes. See revised 3.7.2.1.5, 3.7.2.1.8.3.1, and 3.7.2.7.~~

See attached GE response

update of ~~QUESTION 220.05~~ 130.020
3.7.2

NRC CONCERN

In your response to previous question (Q130.020) you stated that for the NSSS equipment where the response spectrum method of seismic analysis was used the closely-spaced modal responses were combined by the SRSS method. The staff position is that for closely-spaced modes, rules set forth in Regulatory Guide 1.92 should be used. Accordingly, state your intent to comply with the position or provide justification and assess the impact for the deviation.

NSSS RESPONSE

The design basis for WNP-2 with regard to the combination of closely-spaced modes is SRSS prior to the issuance of Reg. Guide 1.92. Subsequently, all NSSS scope of supply has been evaluated and shown to meet the requirements of Reg Guide 1.92. (double sum with absolute sign). Accordingly, the FSAR Section 3.7.2.1.5 is amended as attached to reflect the satisfaction of these requirements.

The maximum modal displacements, $\underline{v}_{r_{max}}$, for the r^{th} mode is:

$$\underline{v}_{r_{max}} = \underline{\phi}_r \frac{\psi_r S_{vr}}{\omega_r} \qquad \text{(Eq. 3.7.2.1-9)}$$

If the design earthquake is specified in terms of a response acceleration spectrum instead of a velocity spectrum, the maximum modal displacements, $\underline{v}_{r_{max}}$, of the structure for the r^{th} mode are:

$$\underline{v}_{r_{max}} = \underline{\phi}_r \frac{\psi_r S_{ar}}{\omega_r^2} \qquad \text{(Eq. 3.7.2.1-10)}$$

hypo

Where: S_{ar} - Spectral acceleration for the r^{th} mode.

The maximum modal inertia forces, $\underline{F}_{r\ max}$, for the r^{th} mode are computed from:

$$\underline{F}_{r_{max}} = \underline{k}\ \underline{v}_{r_{max}} \qquad \text{(Eq. 3.7.2.1-11)}$$

With maximum modal displacements and modal inertia forces known, the other modal quantities such as shears and moments are computed for each mode by conventional structural analysis procedures.

The individual modal maxima are combined by the square root of the sum of the squares method. For example, the total displacements of the structure $\underline{v}_{tot}$ are computed from:

$$\underline{v}_{tot} = \sqrt{\Sigma \underline{v}^2_{r_{max}}} = \sqrt{\underline{v}^2_{1_{max}} + \underline{v}^2_{2_{max}} \cdots + \underline{v}^2_{n_{max}}} \qquad \text{(Eq. 3.7.2.1-12)}$$

The total internal forces, such as shears and moments, are computed in the same manner.

3.7.2.1.5.1 Combination of Modal Response

In a response spectrum modal dynamic analysis, if the modes are not closely spaced (i.e., if the frequencies differ from each other by more than 10 percent of the lower frequency), the modal responses are combined by the square root of the sum of the squares (SRSS) method as described in Subsection 3.7.2.1.5.1.1. If some or all of the modes are closely spaced, a double sum method, as described in Subsection 3.7.2.1.5.1.2 is used to evaluate the combined response. In a time-history method of dynamic analysis, the vector sum at every step is used to calculate the combined response. The use of the time-history analysis method precludes the need to consider closely spaced modes.

3.7.2.1.5.1.1 Square Root of the Sum of the Squares Method

Mathemetically, this SRSS method is expressed as follows:

$$R = \left(\sum_{i=1}^{n} (R_i)^2 \right)^{1/2} \qquad \text{(Eq. 3.7.2.1-12)}$$

where:

R = Combined Response

R_i = Response in the i^{th} mode

n = Number of Modes considered in the analysis.

3.7.2.1.5.1.2 Double Sum Method

This method is defined mathematically as:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k \; R_s \right| \epsilon_{ks} \right)^{1/2} \quad \text{(Eq. 3.7.2.1-13)}$$

where:

R = Representative maximum value of a particular response of a given element to a given component of excitation

R_k = Peak value of the response of the element due to the k^{th} mode

N = Number of significant modes considered in the modal response combination

R_s = Peak value of the response of the element attributed to s^{th} mode

where:

$$\epsilon_{ks} = \left[1 + \left\{ \frac{(\omega'_k - \omega'_s)}{(\beta'_k \; \omega_k + \beta'_s \; \omega_s)} \right\}^2 \right]^{-1}$$

in which:

$$\omega'_k = \omega_k \left[1 - \beta_k^2 \right]^{1/2}$$

$$\beta'_k = \beta_k + \frac{2}{t_d \omega_k}$$

where ω_k and β_k are the modal frequency and the damping ratio in the kth mode, respectively, and t_d is the duration of the earthquake.

~~If several controlling frequencies in an eigenvalue solution for non-NSSS piping, equipment, and structures are found to lie close together, their modal maxima are combined by direct summation (sum of absolute values method) and then combined by the square-root-of-the-sum-of-the-squares method with other individual modal maxima. For NSSS equipment and piping systems, responses for closely spaced modes were generally combined by the SRSS method, except that the double-sum method, approved by the NRC on the GESSAR 251 docket, was used to combine the responses for closely spaced modes for the main steam and reactor recirculation ASME Safety Class I piping. Close frequencies are considered those which satisfy the following relationship.~~

$$\omega_r \leq \omega_{r+1} \leq 1.10\omega_r \qquad \text{(Eq. 3.7.2.1-13)}$$

3.7.2.1.6 Time-History Method of Analysis

The time-history of ground acceleration, $v_g(t)$, is defined at discrete time intervals. The acceleration is approximated by a segmentally linear function and the solution to Duhamel's Integral (Equation 3.7.2.1-5) is obtained by using a step-by-step integration procedure (Reference 3.7-7).

Y_r (t) is computed as a function of time for r = 1,2,3,..,n, where n is the number of significant modes of the system. The modal displacements, $\underline{v_r(t)}$, at the time t for the r^{th} mode, are then calculated from:

$$\underline{v_r(t)} = \underline{\phi_r} Y_r(t) \qquad \text{(Eq. 3.7.2.1-14)}$$

The total displacements, $\underline{v(t)}$, of the structure at any time, t, are obtained by adding the individual modal displacements at time t:

$$\underline{v(t)} = \underline{v_1(t)} + \underline{v_2(t)} + \ldots + \underline{v_n(t)} \qquad \text{(Eq. 3.7.2.1-15)}$$

The inertia forces, $F_r(t)$, at time t, for the r^{th} mode are determined from:

$$\underline{F_r(t)} = \underline{k}\ \underline{v_r(t)} \qquad \text{(Eq. 3.7.2.1-16)}$$

NRC - SEB

ISSUE 34

Submit one (1) sample of typical representative calculations for seismic category I underground piping.

Calculations will be sent to NRC by B&R Woodbury February 8, 1982.

Open SER Issue

6.7 Main Steam Isolation Valve Leakage Control System

The MSIV-LCS, as designed, imposes a maximum process load of 80 lbs. of saturated steam at 35 psia, vented to the reactor building volume served by the Standby Gas Treatment System (SGTS), followed by the continuous MSIV leakage flow rate of 46 scfh (11.5 scfh at 25 psig per valve).

The initial discharge will have no significant effect on building pressure buildup and the continuous flow is considered negligible compared to the SGTS design flow rate of 4000 scfm. The MSIV-LPS conditions the exhaust temperature and humidity to within the design requirements of the SGTS prior to delivery to the SGTS by diluting the 46 scfh MSIV Leakage with 50 scfm of air from the reactor building.

WNP-2 Technical Specifications, Section 3.6.1.2, limits the MSIV leakage to 11.5 scfh, per valve, at 25 psig and requires testing each valve to verify the leak rate at least once every 18 months. The MSIV leak rates are, therefore, verified to be within the design capability on a routine basis.

addition

Leakage in the MSIV System is designed to conform with the leakage rate specified in the Standard Tech Specs.

Q. 010.055
(9.1.2)

Describe, discuss, and verify that the maximum potential kinetic energy contained in all objects of less weight than a spent fuel assembly which will be handled over spent fuel will not exceed the effects of the fuel handling accident described in section 15.7.4 of the FSAR.

Response:

Add at the end of ~~Paragraph~~ Section 9.1.2.3.2 "Section 15.7.4 of the FSAR presents an analysis (for radiological considerations) of how many fuel rods would fail in the event a fuel bundle (700# including channel) was accidently dropped on the core. The worst case accident for this scenario would cause 124 fuel rods to fail based on 250 ft. lbs. of energy being required to cause compressive cladding failure in a single rod. The total kinetic energy required to cause the failure of the 124 fuel rods would be 124 times 250 ft. lbs. or 31,000 ft. lbs. This total kinetic energy was used as the limit for the spent fuel pool accidental drop. Loads were calculated that if dropped from (1) the pool surface, (b) 5 feet above the surface, and (c) 20 feet above the pool surface would yield 31,000 ft. lbs. on impact with the spent fuel assemblies being stored in the high density racks. The fuel assemblies in the rack would be the first structure encountered by a falling object since the BWR assemblies protrude above the racks by approximately 4½ inches. The maximum weights of objects dropped from the three aforementioned conditions are 1510 lbs, 1210 lbs, and 760 lbs respectively. The maximum weight of 1510 lbs was chosen as the maximum allowable load that would be handled over the spent fuel pool surface. While the 1510 lbs is slightly greater than twice the weight of a fuel assembly, this value was chosen as a maximum credible number for analysis purposes concerning criticality and structural aspects of the fuel racks. Any loads less than this value dropped from the same height would yield a lower kinetic energy value and therefore result in less impact on fuel assemblies being stored in the fuel racks. Since the kinetic energy due to the maximum weight of 1510 lbs is equal to that considered in the fuel handling accident of Section 15.7.4 of the FSAR, any kinetic energy less than this amount being dissipated will result in a lesser amount of cladding failures in fuel rods. Therefore, the effects from dropping of an object of less weight than a spent fuel assembly which is being handled over the surface over the spent fuel pool will be less than that described in 15.7.4.

The high density fuel rack designer, NUS Corporation, analyzed the fuel racks from both a structural and criticality standpoint concerning a 1510 lb object dropped from the surface of the fuel pool. The results indicated that noe of the fuel rack damage that might occur in this situation would lead to a criticality problem.

The details of these analyses are given in NUS Corporation Technical reports, 5326-FA-01 and G-RA-17 entitled, "Structural Analysis of the WNP-2 Rack and Fuel Assemblies for an Accidental Object Drop Loading Condition," and, "Criticality Analysis of Dropped Object Accident for WNP-2 Spent Fuel Storage Rack," respectively. FSAR page changes attached.

Tables 010.055-1 and 010.055-2 have been prepared as a response to this question. (see attached)

TABLE 010.055-1

LIGHT LOADS OVER THE SPENT FUEL POOL

Item	Distance Above Pool Surface or Above Fuel Rack in Pool (in feet) Above Pool	Above Rack	Weight No.	Kinetic Energy at Impact (top of rack) ft/#
1. Channel Bolt Wrench	4		40	982
2. Channel Handling Tool	14		75	2,592
3. Channel Guaging Fixture		13*	210	2,389
4. General Purpose Grapple	14		25	864
5. Clam Shell Retriever	4		14	344
6. Manipulator Grapple	4		50	1,228
7. Actuating Pole	3#	20.5*	100	2,075
8. General Area Underwater Light	4		40	982
9. In Core Detector Cutter	4		150	3,684
10. Fuel Support Grapple	4		147	3,610
11. Peripheral Orifice Grapple	4		45	1,105
12. Peripheral Orifice Holder	4		130	3,193
13. Blade Guide	3*	20.5*	170	3,527
14. Fuel Bail Cleaner	4		100	2,456
15. Grid Guide	3#	20.5*	175	3,631

* Distance of CG to top of rack.
\# Distance of CG above pool.

TABLE 010.055-1

LIGHT LOADS OVER THE SPENT FUEL POOL

16. Dummy Fuel Assembly		8.2	600	4,305
17. Peripheral Fuel Support Plug	4		300	7,368
18. Fuel Grapple		23.5	100	2,056
19. Control Tube Grapple	4		45	1,105
20. Guide Tube Grapple	4		35	860
21. Control Rod Latch Tool	4		45	1,105
22. Fuel Bundle Sampler	4		650	15,964
23. Fuel Bundle & Channel		8.2	697	5,001
24. Fuel Bundle & Channel w/Grapple		8.2/15/7	697/100	6,375

* Distance of CG to top of rack.
\# Distance of CG above pool.

Note 1: Assumed to be fuel grapple.

TABLE 010.055-2

LIGHT LOADS OVER REACTOR VESSEL CORE

ITEM	WEIGHT OF ITEM	HEIGHT IN AIR	POTENTIAL ENERGY IN AIR FT.-LB.	.875 x HEIGHT IN WATER	POTENTIAL ENERGY WATER FT.-LB.	TOTAL POTENTIAL
General Purpose Grapple	25#	6'	150	46'	1,150	1,300
Manipulator Grapple	50	6	300	46	2,300	2,600
J-Hook or L-Hook with 6 Sections of Pole + 5 Connectors & 1 Tee Handle	47	6	282	46	2,162	2,444
Rail Clamp	3	29	87	27	81	168
Clam Shell Retriever	15	6	90	46	690	780
Magnetic Retriever	2	6	12	46	92	104
General Area Underwater Light	40	29	1,160	27	1,080	2,240
Local Area Underwater Light	20	29	580	27	540	1,120
Drop Light	25	29	725	27	675	1,400
Underwater TV	25	29	725	27	675	1,400
Viewing Aid	11	6	66	46	506	572
Light Support Bracket	70	29	2,030	27	1,890	3,920
Fuel Support Grapple	87	6	522	46	4,002	4,524
Instrument Strongback	600	29	17,400	27	16,200	33,600
Peripheral Orifice Grapple	45	6	270	46	2,070	2,340
CRD Guide Tube Seal	150	29	4,350	27	4,050	8,400
In Core Guide Tube Seal	120	29	3,480	27	3,240	6,720
Peripheral Orifice Holder	130	29	3,770	27	3,510	7,280
Blade Guide	170	6	1,020	46	7,820	8,840
Fuel Bail Cleaner	100	6	600	46	4,600	5,200
Grid Guide	175	6	1,050	46	8,050	9,100
Dummy Fuel Assembly	600	6	3,600	46	27,600	31,200
Fuel Grapple	1,000	6	6,000	46	46,000	52,000
In Vessel Storage Rack	575	29	16,675	27	15,525	32,200
Control Rod Grapple	45	29	1,305	27	1,215	2,520
CRD Guide Tube Grapple	35	29	1,015	27	945	1,960

WNP-2

TABLE 010.055-2

LIGHT LOADS OVER REACTOR VESSEL CORE

ITEM	WEIGHT OF ITEM	HEIGHT IN AIR	POTENTIAL ENERGY IN AIR FT.-LB.	.875 x HEIGHT IN WATER	POTENTIAL ENERGY WATER FT.-LB.	TOTAL POTENTIAL
Stud Handling Tool	135	29	3,915	27	3,645	7,560
RPV Stud	480	29	13,920	27	12,960	26,880
Shroud Head Butt Wrench	110	29	3,190	27	2,970	6,160
Head Stud Rack	300	29	8,700	27	8,100	16,800
Steamline Plug & Installing Tool	500	29	14,500	27	13,500	28,000

was performed using the horizontal floor response spectra (damping 1/2% of critical).

The fundamental frequency of vertical vibration of the rack was also determined using the STARDYNE computer program. The same model replacing lateral mass with vertical mass was utilized. In this case, since the fuel rests on the base frame, the entire mass of the fuel was lumped at the base grid. Since the calculated frequency was 50.9 Hz and the vertical floor response spectra (damping 1/2% of critical) showed constant acceleration at frequencies in excess of 18 Hz, the effects of the vertical accelerations were considered using the zero-period acceleration in a static analysis. The lateral and vertical loads were considered to be acting simultaneously.

In the general seismic/structural analysis of the fuel racks, the mass of a fuel assembly is assumed to be uniformly distributed along the length of each of the fuel storage cans. Since a maximum gap on the order of 3/8" exists between the side of a fuel assembly and the can (when the fuel is not encased in a channel), the fuel will actually move within the can during a seismic event and cause impact loads to be transmitted to the fuel rack restraints. The effects of this fuel can interaction are determined using a simplified finite element model of the rack and fuel. A nonlinear dynamic analysis is performed utilizing the ANSYS computer program. Details of this analysis are given in NUS Corporation Technical Report #2060, entitled "Fuel-Can Interaction Analysis," October, 1977.

Using the given loads, load combinations and analytical methods, stresses were calculated at critical sections of the rack and compared to the structural acceptance criteria. In all cases, the calculated stress did not exceed the allowable stress.

To assure the integrity of the spent fuel storage racks, specially designed control samples, consisting of B_4C plates sealed in storage tube stainless steel material and fabricated using the same procedures employed for the production of the fuel racks, will be placed in a readily accessible position in the spent fuel pool. These samples are subjected to periodic visual examination and neutron attenuation tests, if visual examination indicates evidence of corrosion.

Insert attached

Insert to Page 9.1-15

Add at the end of ~~Paragraph~~ Section 9.1.2.3.2 Section 15.7.4 of the FSAR presents an analysis (for radiological considerations) of how many fuel rods would fail in the event a fuel bundle (700# including channel) was accidently dropped on the core. The worst case accident for this scenario would cause 124 fuel rods to fail based on 250 ft. lbs. of energy being required to cause compressive cladding failure in a single rod. The total kinetic energy required to cause the failure of the 124 fuel rods would be 124 times 250 ft. lbs. or 31,000 ft. lbs. This total kinetic energy was used as the limit for the spent fuel pool accidental drop. Loads were calculated that if dropped from (1) the pool surface, (b) 5 feet above the surface, and (c) 20 feet above the pool surface would yield 31,000 ft. lbs. on impact with the spent fuel assemblies being stored in the high density racks. The fuel assemblies in the rack would be the first structure encountered by a falling object since the BWR assemblies protrude above the racks by approximately 4½ inches. The maximum weights of objects dropped from the three aforementioned conditions are 1510 lbs, 1210 lbs, and 760 lbs respectively. The maximum weight of 1510 lbs was chosen as the maximum allowable load that would be handled over the spent fuel pool surface. While the 1510 lbs is slightly greater than twice the weight of a fuel assembly, this value was chosen as a maximum credible number for analysis purposes concerning criticality and structural aspects of the fuel racks. Any loads less than this value dropped from the same height would yield a lower kinetic energy value and therefore result in less impact on fuel assemblies being stored in the fuel racks. Since the kinetic energy due to the maximum weight of 1510 lbs is equal to that considered in the fuel handling accident of Section 15.7.4 of the FSAR, any kinetic energy less than this amount being dissipated will result in a lesser amount of cladding failures in fuel rods. Therefore, the effects from dropping of an object of less weight than a spent fuel assembly which is being handled over the surface over the spent fuel pool will be less than that described in 15.7.4.

The high density fuel rack designer, NUS Corporation, analyzed the fuel racks from both a structural and criticality standpoint concerning a 1510 lbs object dropped from the surface of the fuel pool. The results indicated that none of the fuel rack damage that might occur in this situation would lead to a criticality problem.

The details of these analyses are given in NUS Corporation Technical reports, 5326-FA-01 and G-RA-17 entitled, "Structural Analysis of the WNP-2 Rack and Fuel Assemblies for an Accidental Object Drop Loading Condition," and "Criticality Analysis of Dropped Object Accident for WNP-2 Spent Fuel Storage Rack," respectively.

~~Table 010.055 has been prepared as a response to this question. (See attached).~~

Response to Q. 040.86

WNP-2 commits to the installation of an air dryer assembly to be installed between the air receiver tanks and the air compressor. In addition, appropriate maintenance and surveillance procedures will be developd to ensure proper operation of the assemblies. This assembly will be installed ~~by the end of the first refueling.~~

before fuel load.

Q. 40.46
(9.5.3)

Identify the vital areas and the hazardous (e.g., high radiation) areas where emergency lighting is needed for safe shutdown of the reactor and the evacuation of personnel in the event of an accident, including fires. Provide a tabulation of these emergency lighting systems in the WNP-2 facility.

Response:

a. Those areas identified in the answer to question 40.45 (part a.) are provided with emergency lighting to facilitate communications and equipment operation which may be needed for safe shutdown of the plant in the event of an accident.

b. Those "radiation areas" and "high radiation areas" depicted by the colors blue, orange, and red on the Radiation Zone Drawings contained in the FSAR, Chapter 12.3, Figures 12.3-1 through 12.3-6, are provided with emergency lighting to facilitate evacuation of personnel in the event of an accident or fire.

c. In addition, the remainder of those vital areas, not included in items a. and b. above, i.e., the main control room, the access route to the remote shutdown room, and the remote shutdown room are provided with various emergency lighting systems as described in the FSAR, 9.5.3.2.

See Table 9.5-8 for the locations of the various plant emergency lighting systems. Definitions of these lighting systems are given in Sections 9.5.3.2.1, 2, 3, and 4.

The table has been modified to show that there is emergency battery powered lighting in the D.G. corridor. Emergency lighting fed from the diesels provided in the 4160 switchgear rooms. The Supply System feels emergency lighting is not necessary in the RHR valve rooms. Everything in these rooms is operable from the control room. There are no valves in these rooms which would require manual cycling in an emergency loss of power situation.

Flourescent or H.I.D. sources are not used inside primary containment; incandescent sources are used inside primary containment.

9.5.3.2. System Description

The plant lighting system consists of four parts: normal ac lighting, normal-emergency (E) ac lighting, dc lighting, and battery powered emergency lighting.
For Location of various plant emergency lighting systems, see Table 9.5-8.

9.5.3.2.1 Normal AC Lighting Systems

This system consists of two completely redundant systems (A & B) which are energized continuously from the plant non-safety related 480 volt auxiliary system motor control centers directly from 3 phase 480 volt, or through 208Y/120 volt dry type lighting transformers and local area lighting panels. Fluorescent, incandescent, and H.I.D. sources are used for the normal ac lighting system.

9.5.3.2.2 Normal-Emergency (E) Lighting Systems

Normal - emergency (E) lighting is provided for safe and orderly shutdown during the loss of normal ac power. This system is energized continuously from the safety related 480 volt motor control centers through 3 phase, 4 wire 208Y/120 volt dry type lighting transformers. These transformers feed E lighting panels.

This lighting system consists of two completely redundant systems (Divisions 1 and 2). Each system has ac lighting energized continuously from critical buses which are connected both to offsite power sources and associated standby diesel generators. Upon loss of offsite power, each bank of the E lighting load is renergized from its associated standby diesel generator source. The standby diesel generators are installed to Seismic Category I requirements.

E lighting comprises approximately 15 percent of the normal plant lighting load and consists of fluorescent, sodium (outdoor) and incandescent sources. E lighting fixtures in the main control room are designed and supported as Seismic Category I.

TABLE

Locations of Emergency Lighting

	Normal AC	Normal Emergency	DC	Batt.
RADWASTE BLDG.				
Control Room	-	x	x	x
Remote Shutdown Room	-	x	x	x
#1 RPS Room, el. 467	x	x	-	x
Vital 4160v Swgr Sm-7 el.467'	x	x	-	-
Vital 4160v Swgr Sm-8 el. 467'	x	x	-	-
TURBINE BLDG.				
Local Feed Pump Control Stat. el 441'	x	x	-	-
Hotwell Level Control Stat. el 441'	x	-	-	-
NonVital 4160v Swgr el 471'	x	x	-	x
DIESEL GENERATOR BLDG.				
Corridor, el 441'	x	-	-	x
Bldg	x	x	-	x
STANDBY SW BLDG.				
# 1	x	-	-	x
# 2	x	-	-	x
REACTOR BLDG.				
ECCS Equipment, el. 420' & 441'	x	x	-	x
RHR Valve Room #1, el. 471'	x	-	-	-
RHR Valve Room #2, el. 471'	x	-	-	-
Containment Air Compressors, el. 501'	x	-	-	-
Reactor Closed Cooling Pumps, el. 548'	x	x	-	x
Hydrogen Recombiner, el. 572'	x	x	-	-
CIRCULATING WATER PUMPHOUSE	x	-	-	x
MAIN GUARDHOUSE	x	x	x	-

Q. 040.17
(9.5.4)

Indicate in the system description of 9.5.4 and Figure 9.5-4 whether there is an overflow line to return excess fuel oil delivered by the transfer pump back to the fuel oil storage tank. Provide such a line or, alternatively, provide justification for not including an overflow return line in your design.

RESPONSE:

The text of 9.5.4.2 has been revised to incorporate the response to this question.

In addition to the text revision, WNP-2 commits to running the fuel oil transfer pumps with discharge valve closed for a period of one hour during the preoperational phase. This will demonstrate that the pump motor will not overheat when in this configuration.

In each supply subsystem, a transfer pump powered from a UPS bus takes suction from the diesel oil storage tank and discharges to an associated diesel generator fuel oil day tank to maintain the fuel oil level within the day tank. The transfer pump is sized to provide a flow of 4.4 times the maximum engine consumption rate and is automatically controlled by level switches activated by day tank fuel level. The capacity of each fuel oil storage tank is sufficient to provide 7 days of operation for the diesel generator being served.

Each transfer pump is connected to a day tank. There is a pipe interconnecting the transfer lines. By shutting off the isolation valves in the cross interconnecting line, the fuel oil can be pumped from storage tank "A" and from storage tank "B" to day tank "B". By shutting off the isolation valves at day tank "A" and at transfer pump "B" and opening of the isolation valves is the cross connection line the fuel oil can be pumped from storage tank "A" to day tank "B". By the same logic the fuel oil can be pumped from storage tank "B" to day tank "A". If a rupture occurs in the transfer line between one storage tank and its associated day tank, the interconnecting cross line will be isolated by shutting off the isolation valves in that line. This will assure adequate fuel oil supply to the other day tank. If a rupture occurs in the interconnecting cross line, this line will be isolated and thereby the fuel oil supply will not be interrupted between any storage tank and its associated day tank.

The volume of the day tanks permits eight and one-half hours of engine operation of the associated diesel generator without resupply to the day tank. This arrangement provides three hours of operation before the transfer pumps starts, two hours of operation between a start signal to the transfer pump and day tank low level alarm in the event the transfer pump does not start, and three and one-half more hours of operation after low level alarms are actuated to take required corrective action.

At normal high oil level a switch will shut off the pump. If the fuel oil level reaches two inches above the normal high level, a high high condition is reached and a second level switch is activated. This switch will close the solenoid valve at the day tank inlet~~, trip the transfer pump~~ and send an alarm signal. Any excess oil will return to the fuel oil storage tank through the one half inch minimum flow line.

Operation of the fuel storage tank transfer pump is controlled manually when fuel is being transferred through the interconnecting line from storage tank "A" to day tank "B" or from storage tank "B" to day tank "A". High and low level annunciation of the day fuel levels will prevent overfilling or depleting the day tank when the transfer pump is on manual control.

The fuel oil supply from the day tanks to each diesel engine being served consists of two mutually redundant systems. Either system is capable of supplying fuel oil to the engine fuel oil to the engine fuel header. Each system contains a fuel supply line, strainer, fuel oil pump, duplex filter, pressure gage, and relief and check valves. Separate fuel return lines from the relief valves to the day tanks are provided for each system on diesel generators 1A and 1B. The HPCS diesel utilizes a common return line to the day tank.

One of the fuel supply pumps is mechanically driven by the engine and is normally used during engine operation. The other supply pump is driven by a 120 volt d-c motor and is used to fill the fuel oil system and fuel header prior to initial operation and after maintenance has been performed on system piping and components. The motor driven pump is also available for engine operation in the event fuel supply through the engine driven pump system fails.

The fuel pumps are located 2.3 feet higher than the inlet suction pipes inside the day tank. The fuel pumps are designed to operate at a slight negative suction pressure.

The fuel oil supply and return piping is not exposed to ignition sources such as open flames or hot surfaces. The transfer lines between the storage and day tanks are buried and the lines between the day tanks and engines are routed through trenches in the diesel generator rooms.

The fuel oil day tank is located in a separate ventilated room which is sized to contain the full contents of the tank should a leak develop. For discussion of fire protection see 9.5.1.

The fuel oil storage tanks are provided with individual and fill air vent lines. If a vent of fill line is ruptured due to tornado or turbine missiles, one storage tank can supply both diesel generators 1A and 1B. The seven day operation

Q. 040.32
(10.2)

Provide a discussion on the inservice inspection program for throttle-stop, control, reheat stop and interceptor steam valves and the capability for testing essential components during turbine-generator system operation.

RESPONSE:

The text of 10.2.2 and 10.3.4 has been revised to incorporate the response to this question.

We will perform surveillance on these valves per technical specification surveillance requirement 4.3.8.2 (Tech Spec Rev 4).

INSTRUMENTATION

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM (Optional)

LIMITING CONDITION FOR OPERATION

3.3.8 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one turbine ~~control~~ governor valve, or one turbine throttle ~~stop~~ valve ~~or one turbine reheat stop valve~~ per high pressure turbine steam lead inoperable and/or with one turbine reheat stop valve or one turbine interceptor valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours or close at least one valve in the affected steam lead(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

4.3.8.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each of the following valves through at least one complete cycle from the running position (~~for the~~ overspeed protection control system)

~~a) For the overspeed protection control system;~~

~~1) Four high pressure turbine control valves, and~~

~~2) Four low pressure turbine interceptor valves~~

~~b)~~ For the electrical overspeed trip system and the mechanical overspeed trip system;

1) Four high pressure turbine throttle ~~stop~~ valves,

2) ~~Four high~~ Six low pressure turbine reheat stop valves,

3) Four high pressure turbine ~~control~~ governor valves, and

4) ~~Four~~ low pressure turbine interceptor valves.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by direct observation of the movement of each of the above valves through at least one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION of the turbine overspeed protection instrumentation.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or corrosion.