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SUBJECT: Forwards response to NRC 931229 RAI re power uprate,w/regard to increase of bottom head pressure & high pressure scram setpoint on structural & functional integrity of control rod drive sys.

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May 13, 1994
G02-94-115

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION,
POWER UPRATE REVIEW (TAC No. M87076)

Reference: Letter, dated December 29, 1993, JW Clifford (NRC) to JV Parrish (SS), "Request for Additional Information, Power Uprate Review (TAC No. M87076)"

The attached document provides the response to the request for additional information included with the reference.

Sincerely,

JV Parrish

J.V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

WCW/bk
Attachments

cc: LJ Callan - NRC RIV
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JW Clifford - NRC
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NRC Site Inspector - 927N

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REQUEST FOR ADDITIONAL INFORMATION,
POWER UPRATE REVIEW (TAC NO. M87076)

1. *(Section 2.5.1) The evaluation did not address the effects of the power uprate such as the increase of the bottom head pressure and the high pressure scram setpoint, on the structural and functional integrity of the control rod drive system (CRDS). Please state the basis for determining the acceptability of the CRDS regarding compliance with the design code. The information provided should include the code edition, the code allowables, the calculated maximum stresses, deformation, and fatigue usage factor for the uprated power conditions, and assumptions used in the calculations.*

RESPONSE

The WNP-2 Control Rod Drive (CRD) system was evaluated for a bounding reactor dome pressure of 1060 psig and an additional 35 psid for the vessel bottom head. The CRD mechanism structural and functional integrity was deemed acceptable for the vessel bottom head pressure of 1095 psig. This bounds the uprated dome pressure of 1020 psig.

The components of the CRD mechanism designated as primary containment pressure boundary have been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The applicable ASME Code effective date for the WNP-2 CRDs is 1968 edition up to and including winter 1970 addenda. The limiting component of the CRD mechanism is the indicator tube which has a calculated stress of 20,790 psi (allowable is 26,000 psi). The maximum stress is due to a maximum CRD internal hydraulic pressure of 1750 psig. The analysis for cyclic operation was performed in accordance with ASME Code, N415.1 (NB-3222.4(d)). All of the requirements of N415.1 (NB-3222.4(d)) are satisfied even when considering the increased power uprate vessel bottom head pressure and high pressure scram set point, thereby satisfying the peak stress intensity limits governed by fatigue. It should be noted that the CRD has been successfully tested for all operational modes at simulated reactor vessel pressures up to 1250 psig saturated conditions. Additional analysis shows that the maximum usage factor calculated per NB-3222.4(d) is 0.15 for the CRD main flange which is less than the allowable limit of 1.0. In view of the fact that there exists adequate stress margins and successful testing, it is concluded that deformations are negligible and not a concern at power uprate conditions.

The CRD system is capable of providing 250 psi differential pressure between the hydraulic control unit and the reactor vessel for control rod insert and withdraw operation. When the reactor is close to or at fully operating pressure, reactor pressure alone will insert the control rod in the required time, although the accumulator does provide additional margin at the beginning of the scram stroke. Therefore, the CRD will perform all its safety function operations at WNP-2 power uprate conditions.

2. *(Section 2.5.1) This section states, "The flow required for CRD cooling and driving are assured by the automatic opening of the system flow control valve, thus compensating for small increases in reactor pressure. Prior to implementation of power uprate the flow control valves will be assessed to ensure that they are capable of operating within their acceptable range." Provide a discussion covering the licensee's planned action if the flow control valves are not capable of operating within an acceptable range and information on similar CRD systems, of the WNP-2 vintage, that have been tested or operated under the uprate system flow and dome pressure.*

RESPONSE

There are standard procedures which dictate the removal and refurbishment of those FCVs which, in rare instances, fail to perform within the acceptable range. In most cases, this step will correct the problem. However, if refurbishing the valves does not result in them performing properly, they can be replaced with new valves.

Fermi-2 and WNP-2 have similarly designed CRD systems. Fermi-2 was licensed and operated at power uprate conditions corresponding to a dome pressure of 1030 psig which is 10 psi higher than that of the WNP-2 power uprate condition.

3. *(Section 3.3.2) It appears that the evaluation of reactor internals did not address the code and edition used for evaluating stresses and allowables for the reactor vessel and internals. Please provide such information and list the maximum stresses, fatigue usage factor and location of highest stressed areas for both the current design and the uprated power conditions.*

RESPONSE

The evaluation of the reactor internals uses ASME Boiler and Pressure Vessel (B&PV) Code Section III, Subsection NB as a guide for design acceptance criteria. The specific applicable Code edition for the reactor pressure vessel, including the shroud support, is the 1971 edition with addenda up to and including the summer 1971 addenda.

The stresses or loads for 16 reactor components were evaluated by either confirming that uprate load combinations are bounded by previous analyses or by scaling stresses using conservative load ratios from these analyses. In some cases, previous analyses were repeated as required to demonstrate acceptance. The structural integrity of the component is demonstrated by comparison with applicable allowable stresses. The stress results for three of the highest stress components are shown in Table 1. For power uprate conditions, the highest reactor internals usage factor of 0.696 was calculated for the reactor pressure vessel feedwater nozzle thermal sleeve.

TABLE 1

COMPARISON OF MAXIMUM STRESSES AND LOCATIONS FOR REACTOR INTERNALS AT CURRENT AND POWER UPRATE CONDITIONS

A. UPSET CONDITION

COMPONENT	MAXIMUM STRESS LOCATION[1]	MAXIMUM STRESS COMPARISON[2] CURRENT	MAXIMUM STRESS COMPARISON[2] UPRATED
Top Guide[3]	Longest Beam	28.5 ksi (31.7 ksi)	28.5 ksi (31.7 ksi)
Shroud	Top Guide Wedge to Shroud Junction	14.6 ksi (21.5 ksi)	14.6 ksi (21.5 ksi)
In-core Housing/ Guide Tube	Tube Section	25.2 ksi (25.4 ksi)	25.2 (25.4 ksi)

B. FAULTED CONDITION

COMPONENT	MAXIMUM STRESS LOCATION[1]	MAXIMUM STRESS COMPARISON[2] CURRENT	MAXIMUM STRESS COMPARISON[2] UPRATED
Top Guide	Longest Beam	41.1 ksi (50.7 ksi)	40.4 ksi (50.7 ksi)
Shroud	Top Guide Wedge to Shroud Junction	16.1 ksi (42.9 ksi)	16.1 ksi (42.9 ksi)
In-core Housing/ Guide Tube	Tube Section	25.8 ksi (39.9 ksi)	25.8 ksi (39.9 ksi)

1. Maximum stress locations are locations where the margin to the allowables is the smallest.

2. Allowable values or acceptance criteria are shown in parenthesis.

3. For Top Guide upset condition, the calculated values are square root sum of squares (SRSS) and the code allowable was increased to account for certified yield strength test data.

4. *(Section 3.3.2) The shroud was evaluated for the changes in reactor internal pressure difference (RIPD) and acoustic loads resulting from the proposed power level uprate. The licensee states "the shroud stresses due to the power uprate RIPD values are below their allowables; and therefore, the shroud is acceptable for the power uprate." Provide a discussion that includes the effects of the acoustic loads on the shroud's acceptability for the power uprate.*

RESPONSE

For a postulated Loss-Of-Coolant Accident as the result of the recirculation line break, the generated decompression waves will impose lateral loads on the shroud. These loads are referred to as acoustic (impulse) loads. Using the BWR/6 251 inch vessel size plant acoustic loads as the basis, GE has empirically calculated the acoustic loads for the non BWR/6 type plants. The acoustic loads for power uprate have conservatively been estimated to have increased 50% compared to those values which were used prior to the power uprate. The power uprate evaluation was performed by revisiting the same analyses which were performed for the current plant design adequacy evaluation. This evaluation incorporated the updated acoustic loads and correct pressure differential reflecting the power uprate conditions. As reflected in Table 1 of the response to request 3, the analysis results indicate that power uprate does not affect the maximum calculated stress for the most limiting shroud location. Therefore, it is concluded that the shroud is acceptable for the power uprate conditions.

6. *(Section 3.3.2) The evaluation of the jet pumps did not address the jet pump holddown beams. Some recent holddown beam failures have been reported. Provide information and a discussion regarding the holddown beams under the proposed power uprate operating conditions with regard to acceptability of stress levels and fatigue considerations.*

RESPONSE

The beam-bolt assembly's function is to clamp the jet pump inlet-mixer to the riser transition piece. The retained installation preload on the beam-bolt assembly is 22,800 lbs, which exceeds the normal and upset hydraulic and differential pressure loads acting on the inlet-mixer to riser joint. For power uprate conditions, the normal and upset hydraulic and differential pressure loads acting on the inlet mixer to riser joint are approximately 13,700 lbs. The sustained operating stress in the beam is essentially due only to the preload on the beam as long as the preload exceeds the operational hydraulic loads on the joint. Thus the stress in the jet pump beam for power uprate is the same as that in the beam for the current rated power and is not affected by power uprate operation. Because the beam loading is a constant load as opposed to a cyclic load, fatigue is not a consideration.

7. *(Section 3.5) This section states that simplified elastic-plastic methods were implemented for the feedwater nozzle and that the code requirements regarding these methods were met. Please provide a detailed discussion on the methodology, assumptions and compliance with the code including edition, and the code allowables used.*

RESPONSE

The evaluation of the feedwater nozzle at uprated power conditions uses the ASME Boiler and Pressure Vessel Code, Section III, Subsections NB-3222 and NB-3223, 1971 edition with addenda up to and including summer 1971. According to these Subsections, the structural adequacy is met if the maximum primary plus secondary stress intensity at a location on the component is less than 3 S_m of the material. If the 3 S_m limit is not met, then plastic behavior is assumed and the simplified elastic-plastic analysis of the ASME Code paragraph NB-3228.3 of winter 1971 addenda can be used to determine structural adequacy.

The power uprate evaluation showed that even though the 3 S_m limit on the range of primary plus secondary stress intensity was exceeded, the criteria of paragraph NB-3228.3 have been met; namely,

a. The calculated range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses is 59.0 ksi which is below the 3 S_m limit of 69.9 ksi.

b. The calculated fatigue usage factor at the power uprate conditions, taking into account the elastic-plastic factor K_e, is .696 which is below the allowable limit of 1.0.

c The largest calculated cyclic range of thermal stress is 80.3 ksi. This calculation conservatively included the mechanical stress. Since the maximum calculated cyclic range of thermal stress is less than the 97.6 ksi(S_n), no thermal ratcheting effect will be experienced and the material meets the thermal ratcheting requirement.

d. The maximum analysis temperature for the material is 522°F for any stress cycle for the nozzle safe end. This is below the maximum temperature in the table in paragraph NB-3228.3 (i.e., 800°F for Nickel-Chrome-Iron). Thus, the material temperature does not exceed the maximum temperature permitted for the material.

e. The calculated ratio of the material's minimum specified yield strength to the minimum specified ultimate strength is 0.44 which is less than the Code requirement of 0.80.

The evaluation of the feedwater nozzle shows that the Code stress limits are met, thus the structural integrity of the feedwater nozzle is acceptable for the power uprate conditions.

REQUEST FOR ADDITIONAL INFORMATION,
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8. *(Section 3.5) This section states that the piping in the primary reactor coolant pressure boundary (RCPB) systems were evaluated for compliance with ASME code stress criteria and fatigue limits. Interface loads to system components such as valves, RPV nozzles, guides, penetrations and piping suspension devices were also evaluated. Provide a discussion regarding analysis methods, assumptions and compliance with their code of record for normal, upset and faulted conditions. The discussion should include the code and Edition used for evaluating the stresses, displacements and fatigue usage for the power uprate. The evaluation should also include a discussion regarding the acceptability of pipe supports and anchorages.*

RESPONSE

The evaluation of the piping at uprated power conditions in the Primary Reactor Coolant Pressure Boundary (RCPB) systems for compliance with ASME Code was performed as follows:

Existing design basis documents (e.g., design specifications, piping stress reports, etc.) were reviewed to determine the design and analytical basis for RCPB piping systems. The power uprate parameters of RCPB systems were then compared with the existing analytical basis to determine the increases in temperature, pressure and flow due to power uprate.

The ASME Boiler and Pressure Vessel Code Section III, Subsection NB-3600 Code equations 9, (which includes normal, upset and faulted conditions) 10, 12, 13 and 14 were reviewed to determine which equations were impacted by temperature, pressure and flow increases due to power uprate on Class 1 piping systems. Similarly, for Class 2 and 3 piping systems, ASME Code Section III, Subsections NC and ND Code equations 8, 9, 10 and 11 were reviewed.

A parametric study was performed by GE for RCPB systems to determine the percent increases in applicable Code stresses, displacements, cumulative usage factors and pipe interface components (including supports) loads as a function of percentage increase in pressure, temperature and flow due to power uprate.

The percent increases were then applied to the highest calculated stresses, displacements and cumulative usage factor at applicable RCPB piping system node points to conservatively determine the "maximum power uprate calculated stresses, displacements and usage factors." This is a conservative approach since power uprate does not affect all of the dynamic loads, e.g., seismic loads are not affected by power uprate.

Finally, these "maximum power uprate calculated stresses, displacements and usage factors" were compared to the Code allowable values to demonstrate Code acceptability.

Similarly, the percentage increase for power uprate conditions from the parametric study was applied to evaluate the pipe supports and anchors of the RCPB piping systems.

The results of these evaluations have shown that the RCPB piping systems satisfy the requirements of the ASME Code Section III, Subsections NB, NC and ND for the power uprate conditions.

9. *(Section 3.5) Please provide an evaluation of the increased SRV discharge dynamic loads and the increased MSIV closure dynamic loads on the main steam line piping. Please also provide an evaluation of the increased fluid dynamic loads on the closure capability of the various safety related valves in the plant.*

RESPONSE

The SRV dynamic loads due to power uprate did not result in exceeding the design values. This is due to the existing margins available in the design loads. Additional information on SRV dynamic loads is discussed in response to request 13.

The increase in the steam flow, due to power uprate, results in higher Turbine Stop Valve (TSV) closure loads which are much higher than the Main Steam Isolation Valve (MSIV) closure loads. Thus TSV loads considered in the main steam piping analysis envelope the MSIV loads. The main steam piping was evaluated for increase in TSV closure loads due to power uprate as discussed in response to request 8.

All containment isolation motor operated valves (MOVs) and other MOVs whose safety function is to close during accident conditions under the plant's GL 89-10 compliance program are being updated to include power uprate conditions. The increased differential pressure loads due to power uprate are included in the plant's GL 89-10 compliance program.

10. *(Section 3.12) State the code and edition used for the power uprate evaluation of balance-of-plant (BOP) piping and pipe supports including anchorages. List the limiting BOP piping systems and components with respect to the maximum stresses and safety margin as a result of the power uprate.*

RESPONSE

The evaluation of the BOP piping systems and supports was performed in a manner similar to the reactor coolant pressure boundary (RCPB) piping system and supports. Please refer to request 8 for a discussion of the analysis method for RCPB.

Tables 2 and 3 depict the stress ratio of projected power uprate stress to Code allowable stress for the BOP piping systems at power uprate conditions.

TABLE 2

BOP CLASS I PIPING STRESS RATIO[1]

SYSTEM	EQ.9	EQ.10	EQ.12	EQ.13	EQ.14 CUM. USAGE FACTOR
FEEDWATER	0.45	1.30	0.51	0.78	0.222
RWCU	0.74	1.28	0.89	0.38	0.588
HPCS	0.71	0.82	Note 2	Note 2	0.049
LPCS	0.64	1.27	0.27	0.82	0.082
RHR	0.67	1.75	0.08	0.94	0.156
SLC	0.81	0.90	Note 2	Note 2	0.014
RCIC	0.64	1.28	0.02	0.83	0.070

Notes:

1. Limiting Stress Ratio is defined as the ratio of power uprate stress to the Code allowable stress.

2. Per ASME B&PV Code Section III, Subsection NB-3600, not applicable since equation 10 was satisfied (i.e., less than 1).

TABLE 3

BOP CLASS II PIPING LIMITING STRESS RATIO[1]

SYSTEM	Equation 8	Equation 9	Equation 10	Equation 11
FEEDWATER	0.72	0.70	0.48	Note 2
RWCU	0.51	0.75	1.14	0.76
HPCS	0.47	0.41	1.12	0.75
RHR	0.54	0.96	0.49	Note 2
SLC	0.37	0.40	0.35	Note 2
RCIC	0.81	0.64	0.57	Note 2
CRD	0.25	0.65	1.02	0.69

Notes:

1 Limiting Stress Ratio is defined as the ratio of power uprate stress to the Code allowable stress.

2. Per ASME B&PV Code Section III, Subsection NC, not applicable since equation 10 was satisfied (i.e., less than one).

11. *(Sections 3.12.1, 10.1 and 10.1.1) The licensee did not provide an evaluation regarding the effects of the power uprate on the design basis analyses of the high energy line break locations, and pipe-whip and jet impingement loads. Please provide such an evaluation which includes methodology and assumptions used in the analyses and the results in comparison with the design basis allowables.*

RESPONSE

Power uprate will not change any of the postulated pipe break locations at WNP-2. A review of the design basis pipe whip and jet impingement calculations was completed. Pipe break events that were originally determined to be "unstable" (pipe end would move or whip) were assumed to remain unstable at the uprated conditions. Break events that were originally determined to be "stable" (limited pipe end movement) were reviewed to ensure they would remain "stable." These stability reviews compared the uprated loading conditions to the original stability criteria. The stable events were concluded to remain stable at the uprated conditions.

The uprated conditions will increase the nominal operating pressure by approximately 1%. The assumed configuration of the jet impingement envelope and the targets identified for each break event are unchanged following the implementation of the power uprate program. However, the jet impingement load is a linear function of the pressure and the impingement loading will also increase by approximately 1%. This slight increase in the jet impingement loading was concluded to have no affect on the original target damage assessment conclusions.

The small increase in the loading will not affect the ability of the pipe whip restraints to perform their design safety function.

The jet impingement loading following implementation of power uprate will increase by approximately 1%. The conclusions reached in the pipe break and jet impingement evaluations are still valid following power uprate for WNP-2.

12. *(Section 3.12.2) Evaluations were performed only for the supports of the systems with the highest loading increases, such as main steam, reactor core isolation cooling (RCIC), recirculation and feedwater systems. Provide a discussion on how the results of these analyses were applied to the other systems which may have smaller safety margins and lower loading increases.*

RESPONSE

The power uprate evaluation of the BOP piping systems was performed by revisiting the existing stress analysis. The comparison of the power uprate parameters to the existing analytical basis indicated that many of the BOP systems were analyzed to conditions that were very close to the conditions expected at the power uprate.

The supports for the systems (main steam, feedwater and recirculation) with the highest loading increases resulting from increase in steam flow, pressure and temperature were evaluated and the results were documented to demonstrate the design adequacy.

The results of the analysis for the systems with highest loading increases were not applied to other systems which may have lower load increases. The remaining systems had negligible load increases (1% or smaller) and were reviewed by random sampling to assure that sufficient margin exists to accommodate the increase due to power uprate increases.

13. *(Section 4.1.2.2) This section states that the SRV containment dynamic loads were reanalyzed for the power uprate conditions and a 3% SRV setpoint tolerance. Please provide a discussion which compares the effects of the SRV loads on various safety related piping and mechanical equipment for the design basis and power uprate conditions.*

RESPONSE

The SRV air-clearing loads include SRV discharge line (SRVDL) loads, suppression pool boundary pressure loads and drag loads on submerged structures. These loads are influenced by SRV opening setpoint pressure, the initial water leg in the SRVDL, SRVDL geometry and suppression pool geometry. Of these, the opening setpoint pressure is the only parametric change introduced by power uprate which can affect the SRV loads. The following evaluations consider the effects of increases in the setpoint pressure with power uprate.

A. SRV Discharge Line and Quencher Loads

The internal SRV discharge line loads and quencher loads for WNP-2 are based on the current SRV setpoints assuming a 1% tolerance. An increase in the setpoints tolerance from 1% to 3% will result in an increase in the internal SRV discharge line and quencher loads. The increase in SRV setpoint tolerance, hence SRV opening pressure, will result in an increase in the SRV flow rate. The internal SRV discharge line and quencher loads are a function of the SRV flow rate. Therefore, the increase in the SRV discharge line and quencher loads can be determined from the increase in the SRV opening pressure. Table 4 depicts the percent increases in the amplitude of the SRV discharge line and quencher loads due to the increase in the opening setpoints for each SRV group at WNP-2.

TABLE 4

INCREASE IN AMPLITUDE OF SRVDL AND QUENCHER LOADS

SRV Group	Current Setpoint +1% Tolerance (psia)	New Setpoint +3% Tolerance (psia)	Increase in SRVDL Loads (%)
I	1176.2	1214.7	3.3
II	1201.5	1224.7	2
III	1211.6	1235.7	2
IV	1221.7	1245.7	2
V	1231.8	1255.7	2

These setpoint increases are a result of both an increase in the tolerance from 1% to 3% and a 15 psi increase in the nominal SRV open setpoint for SRV Group 1 resulting from power uprate. An evaluation was performed which demonstrated that there is adequate margin available between the calculated and allowable stresses in the SRV discharge piping to withstand the effect of the increase in the current setpoints.

B. Pool Boundary and Submerged Structure Loads

The expected increase in SRV loads on the suppression pool boundary and submerged structures in the suppression pool due to changes in SRV setpoints with power uprate will be less than 1%. This is based on GE methods (Reference 1) used for computing pool boundary pressures in Mark II plants which use X-quenchers at the end of the SRV discharge lines. This increase is covered by the margins in the WNP-2 load definition identified in Reference 2 which was reviewed and accepted by the NRC in Reference 3. Therefore, the WNP-2 load definition (Reference 2) for these loads is not impacted by the change in the SRV setpoints and power uprate.

Since the WNP-2 SRV load definition on the suppression pool boundary and submerged structures is not impacted by power uprate, an evaluation of available stress margins is not required for these loads.

References

1. GE Nuclear Energy, "Mark II Containment Dynamic Forcing Functions Information Report," NEDO-21061, November 1981.

2. Burns and Roe Report, "SRV Loads - Improved Definition and Application Methodology for Mark II Containments," July 29, 1980.

3. "Safety Evaluation Report Related to the Operation of the WPPSS Nuclear Project No. 2," NUREG-0892, Supplement No. 1, USNRC, August 1982.

14. *(Section 10.2) This section only discusses the effects of power uprate on the environmental qualification of equipment, but not dynamic qualification. For safety-related equipment, the dynamic qualification should also be addressed with respect to SRV events, annulus pressurization and jet loads in the context of power uprate.*

RESPONSE

The dynamic qualification with respect to SRV events, annulus pressurization (AP) and jet loads is discussed as follows:

As noted in request 13, it was established that the small increase in the SRV loads would not impact the original SRV load defined in Reference 2 of request 13.

Also, as discussed in Section 4.1.2.3 of NEDC-32141P, there was no impact of power uprate on the AP load defined for WNP-2. No other jet impingement loads, beyond the inertial effects of AP are defined for WNP-2 dynamic equipment qualification.

Therefore, the dynamic qualification of safety related equipment with respect to SRV events, annulus pressurization and jet loads is not affected by power uprate.

15. *(Section 10.2.3) This section states that the mechanical component nozzle loads and support loads are not significantly affected by the increased temperature, pressure and flow conditions for power uprate. Please provide an evaluation of the other important aspects of mechanical component design including the increased fluid induced loads on component internals.*

RESPONSE

The effect of fluid induced loads on the safety related system components has been discussed in NEDC-32141P (Sections 3.3, 3.5 and 4.1) and subsequently found that these components meet the appropriate design criteria.

For the non-safety related systems, evaluations were performed to determine whether these systems have adequate capacity at power uprate flow conditions. As part of this evaluation, the expected increase in the fluid velocity through the systems, which directly affects the fluid induced loads, was shown to be within the design practices (e.g., Heat Exchange Institute (HEI)) used for components and piping systems. Six heater drain valves were shown to have insufficient capacities and have been replaced. Velocities in extraction lines for two feedwater heaters exceed normal design practices. Additionally, normal and auxiliary drain inlet nozzle fluid velocities of five feedwater heaters and drain outlet nozzle fluid velocities for all feedwater heaters exceed the maximum velocity recommended by HEI. For those systems which were shown to have velocities that exceed the accepted design practices, appropriate adjustments have been made to the WNP-2 erosion/corrosion test program to monitor impacts such that action can be taken if required. Longer term plans are to replace many feedwater heaters and adjustments to the design will be made at that time.

The effects of increased uprate related temperature, pressure and flow conditions on the turbine, moisture separator and turbine auxiliaries (e.g., turbine inlet features, piping, valves, cross under/over piping and extraction zones, etc.) were evaluated by the turbine vendor, Westinghouse, and determined to meet Westinghouse original design criteria.

16. *(Section 10.3) This section briefly describes the power uprate initial testing program. Please describe in more detail how this program will verify the original licensing requirement to perform tests in accordance with Standard Review Plan Section 3.9.2, especially regarding the increased flow induced dynamic loads.*

RESPONSE

The NRC approved (Reference 1) "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," NEDC-31897P-1, noted that power uprate requires only limited startup tests and a testing plan will be included in the uprate licensing application. The testing described in Standard Review Plan Section 3.9.2 is not required. WNP-2 licensing submittal, NEDC-32141P, Section 10.3 details the plant-specific testing at the time of implementation of power uprate. Based on Section 10.3 commitments, a startup test procedure will be developed for WNP-2.

As noted in the generic guideline (NEDC-31897P-1), "the planned approach to achieve an increase in rated power requires no increase in the maximum core flow." WNP-2 is currently licensed to operate at 106% of the rated core flow. This is the same core flow at which the power uprate evaluation was performed and there is no change in the dynamic loads associated with the recirculation system.

As the result of power uprate, the flow rate through the feedwater system and the main steam lines increases by 5%. Thus, the potential increases in dynamic loads on internals and piping components were evaluated at uprated power (Sections 3.3, 3.5 and 3.12 of NEDC-32141P). The results of these evaluations show that the vessel internals, reactor coolant pressure boundary (e.g., main steam lines), safety/relief valves, SRV discharge lines and feedwater lines have sufficient margin to the design criteria to perform at the analyzed uprated conditions. Therefore, the testing described in Standard Review Plan Section 3.9.2 is not required.

However, in addition to the startup testing based on Section 10.3 commitments discussed above, the Supply System will continue surveillance of various systems in accordance with WNP-2 Technical Specifications during power ascension and steady state operation, as is currently the practice, to ensure proper operation at uprated conditions.

Reference

Letter from W.T. Russell (NRC) to P.W. Marriott (GE), "Staff Position Concerning General Electric Boiling Water Reactor Power Uprate Program (TAC No. 79384)," dated September 30, 1991.

17. *(TS Section 3/4.4.2) The Technical Specification (T/S) tolerance for the safety/relief valves (SRVs) is being revised to +/-3%. However, the T/S Surveillance Requirements for the SRVs should also include a footnote to require that the valves be reset to within +/-1% prior to returning the valves to service. This will provide necessary assurance that the amount of setpoint drift outside the acceptable range is minimized. This is discussed in the licensing topical report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report", as approved by the NRC.*

RESPONSE

The Supply System requests that the changes to Technical Specification 3/4.4.2 be approved as submitted and that the NRC proposed footnote not be imposed as a requirement for the WNP-2 Technical Specification revision. This request is based on the following points which are amplified in the discussion that follows:

- The requested +/-3% tolerance is consistent with the revised ASME tolerance in the ASME OM-1-1987 Code for SRV performance testing. The Supply System is committed to adopting ASME OM-1-1987 after December 13, 1994 as required by 10CFR50.55a.

- The ASME Code previously required a +/-1% tolerance unless a greater tolerance is established in the overpressure protection report and the safety valve design specification. The Supply System has had a plant specific set of analyses generated which fully qualifies and establishes the +/-3% tolerance for WNP-2 SRVs.

- WNP-2 has Crosby spring type SRVs and has not experienced significant setpoint drift. For example, in as-found setpoint tests on 38 SRVs in the last five years (five surveillance test periods) there has been only one case in which the setpoint was found to be out of the current +1/-3% tolerance on the high (plus) side of the tolerance range.

- As-found and as-left setpoints on the minus side of the setpoint tolerance range are not a safety concern, since the overpressure protection evaluation uses the high limit setpoint and the lower limit is not pertinent for this analysis.

- The recent WNP-2 plant specific analyses fully qualify the increase to +3% tolerance on the plus side of the tolerance range based on conservative maximum pressure and flow operating condition assumptions for the associated Reactor Power Uprate, as well as conservative SRV setpoint and out-of-service assumptions. The analytical conservatism assures adequate margin to comply with Technical Specifications using the requested +/-3% tolerance for both as-found and as-left conditions. Using the ASME OM-1 Code tolerance (see first bullet above) for as-left conditions is consistent with NRC interpretation in NUREG-1482, paragraph 4.3.7.

- WNP-2 performs almost all setpoint testing, both ASME/Technical Specification surveillance as well as post-maintenance testing, on-line at low reactor power or on decay heat. On-line setpoint testing of SRVs allows tests to be done at normal system pressure and temperature and at actual drywell ambient conditions in which the SRVs normally operate. Imposition of a new requirement to reset SRVs to +/-1%, even if the SRVs met the required +/-3% tolerance, prior to returning the SRVs to service would require multiple startups and shutdowns after refueling outages in order to meet the requirement. Repeated startup/shutdown cycles are detrimental to the plant as a whole and to plant equipment, more so than the benefit to be gained by resetting one or more SRVs, which otherwise meet Technical Specifications, to a more restrictive +/-1% range.

- The footnote proposed by the NRC would impose a severe penalty on WNP-2 resulting from either abandonment of on-line testing as it is now performed, which would have a substantial loss of investment in equipment and installation made for this purpose, or by requiring detrimental repeated plant startup/shutdown cycles and thereby causing extension of refueling outages. Previous SRV performance and recent analyses do not indicate that this severe penalty is necessary.

The Supply System was not a member of the BWROG committee that generated licensing topical report NEDC-31753P, and thus did not have access to the report. Since the Supply System does not have access to NEDC-31753P, the basis for retaining the previous +/-1% tolerance prior to returning the valves to service is not known. However, it appears that this reference to a +/-1% tolerance is a holdover from the previous ASME code requirement (Subsection NB-7512) which required the +/-1% tolerance unless a greater tolerance is established in the overpressure protection report and the safety valve design specification.

The Supply System requests that the Staff approve the proposed Technical Specification change without the suggested footnote for the following reasons:

1) The requested tolerance (+/-3%) is consistent with the soon to be adopted ASME Code for SRV testing,

2) a new plant specific analysis for WNP-2 supports the +/-3% tolerance,

3) the WNP-2 performance history of out of tolerance high occurrences is low, and

4) significant hardship would be imposed on the present methods of testing the valves if a reset requirement to +/-1% was required.

REQUEST FOR ADDITIONAL INFORMATION,
POWER UPRATE REVIEW (TAC NO. M87076)

The following paragraphs provide an expanded discussion of the Supply System proposal to leave the SRVs undisturbed if they are within the +/-3% tolerance.

Consistent with the ASME Code, the current WNP-2 Technical Specification Section 3/4.4.2 allows a greater setpoint tolerance of +1%/-3%, which is required for operability of the safety function of at least 12 of the 18 installed SRVs, and is based on the valve design specification and the WNP-2 Overpressure Protection Report. The new licensing submittal request to increase the tolerance range to +/-3% is consistent with revised ASME OM-1 Code requirements which allows the increase in the tolerance to +/-3%. The Supply System is committed to adopting ASME OM-1-1987 after December 13, 1994 as required by 10CFR50.55a.

In addition, WNP-2 has generated a new set of analyses (GE-NE 187-24-0992, "Washington Public Power Supply System Nuclear Project 2 SRV Setpoint Tolerance and Out-of-Service Analysis"), which fully qualifies and establishes the increased +/-3% tolerance for the higher reactor power uprate maximum assumed operating conditions and, again, for the condition of at least 12 of the 18 SRVs operable for the SRV safety function (operability includes setpoints within the required tolerance range). These new analyses are <u>plant specific</u> for WNP-2.

It also appears that the reference to a +/-1% tolerance is based on concern for the possibility that the setpoint of a particular SRV may drift outside the new +/-3% tolerance range. With respect to the general concern for setpoint drift, WNP-2 has Crosby spring type SRVs (compared to pilot operated SRVs) and has not experienced significant setpoint drift. For example, in as-found setpoint tests on 38 SRVs in the last five years (five surveillance test periods), there has been only one case in which the setpoint was found to be out of the current +1/-3% tolerance on the high (plus) side of the tolerance range.

With respect to potential setpoint drift of a particular SRV, the drift, should it occur, could be either trending toward the low (minus) side of the tolerance or toward the high (plus) side of the tolerance. Setpoint drift of a particular SRV on the low (minus) side of the tolerance range is in the conservative direction. The overpressure protection evaluation considers only the maximum or high limit of the tolerance range of the SRV opening pressure for all SRVs considered to be in-service because a lower opening pressure on the minus side of the tolerance range is less restrictive and not pertinent to overpressure protection. The safety valve design specification specifies a low limit of 1067 psig (-8.4% to -11.5%) on the SRV setpoint. Therefore, as-found or as-left setpoints on the minus side of the SRV setpoint tolerance range are not a safety concern. This is consistent with ASME OM-1-1981 and OM-1987 Part 1, both as written and as clarified in Interpretation 92-8, in that the code is concerned with the upper limit and an as-found setpoint lower than the lower tolerance limit is not considered a failure of the periodic setpoint test. Thus, setpoints that are outside the existing -3% tolerance (and less than nameplate) are not a safety concern and an even tighter tolerance of -1% prior to return to service is not warranted.

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With respect to potential setpoint drift of a particular SRV on the high (plus) side of the tolerance range, the recent WNP-2 plant specific analyses, as noted above, fully qualifies the new tolerance range of +/-3% based on conservative maximum operating pressure and flow conditions for power uprate, under further conservative conditions of at least 12 of the 18 SRVs operable, and all operable SRVs set at the maximum high tolerance setpoint; this qualifying analysis provides similar safety margins for setpoints up to +3% of nameplate as was provided in the previous analyses for setpoints up to the previous +1% tolerance. As noted, WNP-2 has not had an identified setpoint drift problem, but should an individual SRV(s) experience drift, there is adequate conservatism, in both the assumed plant operating conditions as well as the assumed setpoints and valves out-of-service considerations, to assure Technical Specifications are complied with using the requested +/-3% tolerance for both as-found and as-left conditions. (Note that SRV "operability" includes setpoints within the specified tolerance, and that, analytically, out-of-service or inoperable SRVs are considered to be non-existent for the purpose of mitigating an overpressure transient; whereas when an SRV setpoint out-of-tolerance condition might exist (drift), the SRV would likely still be available to assist in overpressure transient mitigation, depending on the actual setpoint and the severity of the event).

Thus, based on the current analyses and ASME code, it would not be necessary to reset SRVs to a more restrictive +/-1% prior to returning the SRVs to service. This is consistent with NRC interpretation in NUREG-1482, paragraph 4.3.7, as it relates to as-left conditions, to have (at least two consecutive) openings within the code tolerance.

Currently, SRVs requiring maintenance (for example, for seat leakage), other than simple adjustment, are removed from the plant during refueling outages (previously, SRVs were refurbished in-situ, and this is still an option if necessary). SRVs removed from the plant are preferentially replaced if a previously refurbished SRV of the correct nameplate setpoint is available; if the proper replacement SRV is not available, the removed SRV would be refurbished and re-installed in the plant during the outage. SRV maintenance and refurbishment is performed on-site, either during refueling outages, or during non-outage periods if the SRV was replaced during the outage. Only a small fraction of the refurbished SRVs are sent off-site for setpoint testing on a test stand; typically only enough to assure a sufficient number of pre-tested SRVs are on site (4) to allow plant startup for continued testing, and/or those which have undergone full teardown inspections and refurbishments in which there is a high potential that the original setpoint prior to disassembly cannot be preserved after reassembly.

WNP-2 performs almost all SRV setpoint testing on-line, including all ASME/Technical Specification surveillance testing as well as most post-maintenance and post-adjustment setpoint verification testing. The as-found surveillance testing on the code required sample size is normally performed at low reactor power or on decay heat during the reactor shutdown evolution for refueling outages; should an SRV not meet the Technical Specification tolerances, the sample size would be increased per code requirements and testing would continue. An SRV that failed

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the surveillance test, not otherwise requiring maintenance, would be re-adjusted in-situ. Refurbished SRVs (except for a few which may have been pre-tested), and any SRVs which were re-adjusted, are retested on-line at low reactor power during the startup evolution after the refueling outage and must meet Technical Specification requirements.

If a footnote is required to be included in the Technical Specification Surveillance Requirements "to require that the valves be reset to within +/-1% prior to returning the valves to service", as discussed in Question 17, then any SRVs as-found tested on-line during the shutdown that were within the required +/-3% tolerance but not within a +/-1% range, would have to be re-adjusted during the outage and retested on-line during startup; as written, if any of the SRVs being tested during startup (for re-adjustment or post-maintenance) were again within the +/-3% tolerance but not within the +/-1% range, the plant would have to be shutdown and cooled down, the valves re-adjusted and then retested during another startup. Because the re-adjustment process is valve specific and not precisely predictable for any particular valve, repeated cycles of shutdown, cooldown, re-adjustment, startup, and retest, may be needed to satisfy the footnote requirement if it is imposed. Repeated SRV cycling for setpoint tests is detrimental to SRV seat leakage and can lead to even further maintenance/test cycles. Plant startups and shutdowns affect all plant components, not just the SRVs, and impose additional thermal, fatigue and operating cycles on plant equipment. These additional cycles, and the risks associated with additional plant operating manipulations, would be more detrimental to the plant as a whole than the benefit that might be gained for setpoint adjustments on one or more SRVs which would meet Technical Specifications in all other respects but may not meet the more restrictive surveillance footnote if imposed.

If the footnote is imposed as a requirement, the alternative to the repeated startup/shutdown cycles, other than more restrictive setpoint tolerances, would be to abandon on-line setpoint testing and send all SRVs off-site for all required setpoint testing. The Supply System has expended substantial resources, both for the permanently installed setpoint testing equipment on each SRV for on-line testing and for additional spare SRVs, to allow the plant to perform most all SRV setpoint testing on-line and all maintenance on-site, without having to incur the added expense, transportation risks, and potential delays associated with having to rely on off-site testing vendors, and to reduce or minimize outage length. On-line setpoint testing of SRVs also allows tests to be done at normal system pressure and temperature and at actual drywell ambient conditions in which the SRVs normally operate.

Therefore, a requirement to reset SRVs to within +/-1% prior to returning the valve to service would impose a severe penalty on WNP-2 by either not allowing setpoint testing on-line as it is now done, and thereby forcing the abandonment of a substantial investment made for this purpose, or by requiring the plant to perform repeated plant shutdown/startup cycles to re-adjust SRVs to within the +/-1% range, even though the SRVs may meet the other Technical Specification requirements. This does not appear warranted based on previous SRV performance and the recent analyses discussed above.