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Fred Dacimo
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August 3, 2004

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-04-095

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: **Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate (TAC MC1865)**

References:

1. NRC letter to Entergy Nuclear Operations, Inc; "Supplemental Request for Additional Information Regarding Stretch Power Uprate", dated June 30, 2004.
2. Entergy letter to NRC (NL-04-005); "Proposed Changes to Technical Specifications: Stretch Power Uprate Increase of Licensed Thermal Power (3.26%)", dated January 29, 2004.
3. Entergy letter to NRC (NL-04-086); "Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate", dated July 16, 2004.

Dear Sir:

This letter provides additional information, requested by the NRC in Reference 1, regarding the license amendment request submitted by Entergy Nuclear Operations, Inc (Entergy), in Reference 2. The responses are provided in Attachment 1, except that the responses to the fuel design questions were previously transmitted by Reference 3 and the responses to LOCA transient questions 3, 4, and 5 will be provided separately by August 12, 2004.

In addition, Attachments 2 and 3 contain errata pages for the Stretch Power Uprate Licensing Report transmitted in the original license amendment request, Reference 2. A Table summarizing the changes is provided. Attachment 2 pages are for the proprietary version (WCAP-16157-P) and Attachment 3 pages are for the non-proprietary version (WCAP-16157-NP). Since there is no proprietary information on any of these pages, an application for withholding is not required for this transmittal.

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The requested additional information provided in Attachment 1 and the errata pages provided in Attachments 2 and 3 do not alter the conclusions of the no significant hazards evaluation that supports this license amendment request. There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Kevin Kingsley at 914-734-6695.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 3 2004.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

Attachments:
1. Reply to Request for Additional Information
2. Errata pages for WCAP-16157-P
3. Errata pages for WCAP-16157-NP

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ATTACHMENT 1 TO NL-04-095

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING

PROPOSED LICENSE AMENDMENT REQUEST FOR

INDIAN POINT 2 STRETCH POWER UPRATE

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

LOCA Transients

Question 1:

Provide a statement indicating that, prior to operating at the uprated power level, emergency operating procedures will be in place and operator training will be completed to ensure that the actions for switchover to hot leg injection will occur consistent with the stated times.

Response:

As stated on page 10-29 of WCAP-16157-P, training will be implemented prior to the SPU. Revised Emergency Operating Procedure (EOP) changes will be in place and operator training will be completed prior to operating at the uprated power level. This includes EOP changes for switchover to hot leg injection at 6.5 hours, and addition of additional auxiliary feedwater within 10 minutes, and training for these changes.

Question 2:

In Attachment III to the April 12 letter, the licensee stated that new well-mounted dual-element resistance temperature detectors (RTDs) will be inserted into two of the three thermowells and that the third thermowell will be capped for future use.

Provide a justification for the insertion of only two of the three thermowells. Explain if there will be any configuration changes to the current design and if there are any effects on the temperature measurement for the SPU condition.

Response:

There are currently three narrow range RTDs protruding directly (direct immersion RTDs) into each hot leg and three in each loop cold leg. One hot leg and one cold leg RTD in each loop are used to determine the loop average temperature and the loop Delta-T for indication, control and protection circuits. The other two RTD pairs in each loop are installed spares.

For each hot leg and cold leg with the exception of 22 Hot Leg, the existing RTDs will be removed, new thermowells threaded into the bosses, and seal welded. New, well-mounted, dual element RTDs will be inserted into two of the three thermowells. These two thermowells are in the physical locations currently feeding the indication, control and protection circuits. The third thermowell will be plugged and capped for future use if needed. As exists now, one hot leg and one cold leg RTD in each loop will be used resulting in three RTDs in each loop as installed spares (the replacement RTDs are dual element). Thus with the new configuration, there will be one additional available spare in each loop.

For 22 Hot Leg, the existing RTDs are at the 3 o'clock position on the pipe, whereas the other RTDs (21, 23, 24 Hot Leg and 21, 22, 23 & 24 Cold Leg) are all at the 9 o'clock position. To address this issue, 2 new bosses, thermowells, and 2 new (dual element) RTDs will be installed on the opposite side (9 o'clock position) of the 22 Hot Leg pipe. Additionally, one new

LOCA Transients

thermowell and one new dual element RTD will be installed in one of the existing three bosses (3 o'clock position). This will continue to provide temperature monitoring capability from the existing (3 o'clock position) RTD location on 22 Hot Leg. Thus for 22 Hot Leg six RTDs will be installed with 5 spares available.

The response time and uncertainty calculations for the new RTDs have been incorporated in the safety analysis for the SPU. The above configuration change and the new EQ qualified, Class 1E RTDs, will enhance plant reliability, operations and reactor protection capabilities.

Question 3:

The LOCA submittals did not address slot breaks at the top and side of the pipe.
Justify why these breaks are not considered for the IP2 LBLOCA response

Response:

To be provided in future letter.

Question 4:

Provide the LBLOCA analysis results (tables and graphs, as appropriate) to the time that stable and sustained quench is established.

Response:

To be provided in future letter.

Question 5:

Tables 6.2-3 and 6.2.5 in the Application Report provide LBLOCA and SBLOCA analyses results for the IP2 SPU.

Provide all results (peak clad temperature, maximum local oxidation, and total hydrogen generation) for both LBLOCA and SBLOCA. For maximum local oxidation include consideration of both pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation. Also include the results for fuel resident from previous cycles.

Response:

To be provided in future letter.

Nuclear Steam Supply System Fluid Systems

Question 1:

In Section 4.1.7 of the Application Report, the licensee discusses the spent fuel pool (SFP) cooling system. However, the information is only described in general terms and conditions.

Describe the specific methods and controls that will be used to perform the cycle specific calculations required to determine that the SFP cooling system can remove the additional heat load and maintain operating conditions within current design. Are these calculations done in accordance with approved methods?

Response:

The analysis for Spent Fuel Pit (SFP) cooling capacity is known to have significantly conservative assumptions. For example, the analysis assumes that the entire contents of the reactor core is transferred to the SFP instantaneously. In order to proceed expeditiously with core offload activities, an IP2 specific SFP heatup calculation will be performed prior to the outage in accordance with standard Entergy procedures. This calculation will correlate time-after-shutdown to the maximum number of assemblies removed from the core. Each hour, as a presumed quantity of fuel is removed from the core, its decay heat contribution will be added to the SFP. As time progresses, the SFP heat load will increase but will be somewhat diminished due to the normal effects of decay

This correlation is generally presented as a graph or table and is made available to the Control Room staff in charge of authorizing fuel movement in a formal plant procedure. If at any time, it shows that the heat removal capacity of the SFP will be exceeded within the next hour, the Operators will be advised to stand down until such time as offload can proceed.

This solution is similar to an approach that was successfully implemented at Indian Point 3 and is resident in the IP3 UFSAR. The basis of the calculational methodology was the decay heat load algorithm in Branch Technical Position ASB 9-2, from the USNRC Standard Review Plan. While this algorithm may be slightly non-conservative for very old fuel, previously resident in the SFP (as per the methodology of Regulatory Guide 3.54) and since about 95% of the SFP heat load comes from the core offload, the conservatisms inherent in the 9-2 methodology more than compensate for any small non-conservative effects.

Mechanical Equipment Design Transients

Question 1:

Table 3.1-1 of the Application Report compares the design parameters used in the existing design transient development and for the stretch power uprate. The licensee indicated that the current design transients remain bracketing and applicable for the SPU. In addition, these IP2 specific design transients have been used in the NSSS component stress analyses and evaluations presented in Section 5 of this report. The licensee further stated that even though the existing design transients bracket the SPU Program, all of the design transients were redeveloped based on the SPU Program design parameters shown in Table 3.1-1 and re-transmitted to the analysts for use in the IP2 SPU Program.

In light of Table 3.1-1, the cold leg temperature range (between 514.3 to 538.2 °F) appears to be more severe than the current design basis cold leg temperature range. Provide a comparison of the design basis transients used in the current design basis transients and the stretch power uprate conditions for NSSS components stress and fatigue analysis. Clarify how the current design basis transients are applicable for the SPU conditions.

Response:

The cold leg temperature range values noted in this RAI are the Vessel/Core inlet temperature values from Table 2.1-2 (514.3 to 538.2°F). The upper bound on the T_{cold} value used in the existing NSSS design transients for the 1990 Uprate and the MUR uprate bound the value for the uprating (from Table 3.1-1 the T_{cold} for the High T_{avg} case is 547.7°F for the existing plant uprating condition vs. 537.9°F for the SPU Program). These values correspond to the steam generator outlet temperatures in Tables 2.1-1 and 2.1-2. The lower bound on the T_{cold} value for the SPU is lower than the value for the existing design transients but only by 1.5°F (515.5°F for the existing plant uprating Low T_{avg} condition vs. 514.0°F for the SPU Program). A 1.5°F more severe T_{cold} is not of sufficient magnitude to require a design transient revision; there is sufficient conservatism in the existing design transients to encompass a small difference like a 1.5°F difference. As stated in Table 3.1-1 footnote 5, it was left to the component analysts' discretion to use either the 515.5°F existing plant condition T_{cold} or the 514.0°F SPU Program T_{cold} in the fatigue stress analyses.

Piping and Supports

Question 1:

In Section 9.9.3 of the Application Report, the justifications provided on page 9.9-3 for not evaluating the piping and support systems where the increase in temperature, pressure and flow rate are less than 5 percent of the current rated design basis condition are qualitative and nonspecific. For instance, the licensee stated that these increases are some what offset by conservatism in analytical methods used. The licensee also indicated that conservatism may include the enveloping of multiple thermal operating conditions.

Provide the technical basis for not evaluating these piping and support systems. The technical justifications should be based on specific quantitative assessment or intuitively conservative deduction. Also, discuss how the flow effects on the transient loads, which may increase non-proportional to the ratio of flow rate change, are considered (see page 9.9.2).

Response:

All piping systems with change factors greater than 1.0 were evaluated to document pipe stress and support system acceptability. The method of evaluation (i.e., detailed quantitative evaluations or simplified evaluations augmented by field walkdowns) varied based on the piping system and related change factors involved.

The method used to qualify the main steam piping involved detailed computer analysis of the piping system. Although operating temperatures and pressures at SPU conditions were bounded by the existing data considered in the design basis piping evaluations, the main steam piping was evaluated using detailed computer analysis in order to reconcile an approximate 6 percent flow rate increase that results due to SPU conditions. These detailed evaluations were performed to assess the potential increase in fluid transient stresses and loads resulting from a turbine stop valve (TSV) closure event.

A summary of revised main steam system stress levels corresponding to SPU conditions is provided in Table 1. The results presented include existing stress levels (i.e., pre-uprate), revised pipe stress levels for SPU conditions, allowable stress for the applicable loading condition, and the resulting design margin for each piping analysis that was evaluated to reconcile SPU conditions. The design margin provided is based on the ratio of the calculated stress divided by the allowable stress.

Piping and Supports

Table 1 Stress Summary at SPU Conditions

Piping Analysis Description	Loading Condition	Existing Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Design Margin
Main Steam Line 1 (Inside Containment)	DL + LP + TSV	13,230	13,549	19,950	0.68
Main Steam Line 2 (Inside Containment)	DL + LP + TSV	11,709	11,956	19,950	0.60
Main Steam Line 3 (Inside Containment)	DL + LP + TSV	13,317	13,717	19,950	0.69
Main Steam Line 4 (Inside Containment)	DL + LP + TSV	14,251	14,629	19,950	0.73
Main Steam Lines 1, 2, 3 and 4 (Outside Containment)	DL + LP + TSV	18,935	19,789	19,950	0.99

Notes:
1. Loading Condition "DL + LP + TSV" corresponds to the combination of stresses due to deadweight + pressure + turbine stop valve closure effects.
2. Design Margin reported is based on the ration of SPU stress divided by the Allowable Stress

For the remaining piping systems with thermal and pressure change factors greater than 1.0, these piping systems (i.e., condensate, feedwater, extraction steam, feedwater heaters vents and drains, and moisture separator and reheater drains systems) were evaluated using simplified evaluation methods augmented by a field walkdown of the piping systems.

For portions of these piping systems with operating temperatures greater than 150°F, the maximum thermal change factor due to SPU is limited to 1.02 (i.e., a 2 percent thermal expansion increase). For portions of these piping systems with operating temperatures less than or equal to 150°F, the maximum temperature increase is only 1°F.

The maximum pressure increase for any of these piping systems occurs for the feedwater pump discharge piping which experiences an increase from 1183 to 1215 psig (approximately 3 percent).

Although the thermal expansion and pressure increases were very small and were considered not to significantly impact the existing piping system qualification, a turbine building plant walkdown of these piping systems was also performed to review the individual piping layouts and associated pipe support configurations. The purpose of these piping system walkdowns was to assess the adequacy of the installed piping deadweight spans and to review the existing thermal flexibility of the piping systems. The overall assessment from the walkdowns performed concluded that the existing piping that was observed was adequately supported and contained adequate flexibility to accommodate the small pressure and temperature changes resulting from SPU. Piping systems were determined to be adequately supported if the piping was supported by vertical supports, rod hangers or spring hangers, such that piping spans were consistent with

Piping and Supports

the guidance presented in ASA B31.1-1955, Code for Pressure Piping. Piping systems were determined to have adequate flexibility if the following attributes were observed:

- Piping lengths and offsets were consistent with simplified industry methods of determining flexibility (for example, nomographs).
- There were no non-integral or integrally welded piping anchors installed.
- There was a sufficient and reasonable number of piping elbows installed providing thermal flexibility.

The piping and pipe support evaluations concluded that these piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature and pressure effects resulting from SPU conditions.

Generic Issues and Programs

Question 1:

On page 10-22, the licensee indicated that the effect of the SPU on the current pressure locking and thermal binding (PLTB) evaluation of safety-related motor-operated valves (MOVs) and air-operated valves (AOVs) was reviewed. It was determined that the SPU does not introduce any increased challenge for thermal binding and/or pressure locking and does not effect the results and conclusions of the current evaluation.

Provide a summary of the evaluation of SPU effects on PLTB in response to Generic Letter (GL) 95-07 for power-operated valves (POVs) including MOVs and AOVs, with respect to the changes of the parameters such as maximum open and close differential pressure, maximum open and close line pressure, flow rate, fluid, fluid temperature, and ambient temperature, that might affect the valve performance.

Response:

Evaluation of the effect of the SPU on the potential for thermal binding and pressure locking of safety-related power-operated valves in response to GL 95-07 is contained in the response to Generic Issues and Programs, Question 12. This evaluation addresses the effects of process conditions (e.g., fluid temperatures) and ambient temperatures on the potential for PLTB in affected power-operated valves.

Question 2:

On page 10-23, the licensee indicated that an isolated water condition is assumed to exist between 2 MOVs in the return line from loop no. 2 hot leg to the suction of the residual heat removal (RHR) pumps inside containment. The curve for containment temperature as a function of time following a LBLOCA is an input used in the analysis of this piping segment. Due to the relatively small differences between the containment temperature profile used in this analysis and the containment temperature profile for a LBLOCA under SPU conditions, and a greater than 30-percent margin between the calculated maximum pressure and the maximum allowable pressure under Updated Final Safety Analysis Report (UFSAR) criteria, the stresses in this line under SPU conditions continue to remain within UFSAR allowable.

Discuss quantitatively how much the pressure will increase due to the increased temperature for the stretch power uprate since the increase in pipe stress is not linearly proportional to the increase in temperature in the isolated piping segment.

Response:

The current analysis which determines the pressure stresses due to thermal expansion of water in Line No. 10 (14 inches diameter, Schedule 140 piping) following a large break LOCA (LBLOCA) contains two conservative assumptions: (1) the line is considered to be completely free of air, and (2) the line is considered to be full of standing water.

Generic Issues and Programs

Comparison of the containment temperature profile used in the current analysis and the containment temperature profile for a LBLOCA under SPU conditions shows that the containment temperatures used in the current analysis essentially envelope the containment temperatures resulting from the SPU LBLOCA analysis during the temperature rise portion of the transient. The peak temperature in the SPU LBLOCA analysis is approximately 3 degrees higher than the peak temperature in the current analysis.

The results of the current analysis show that the maximum pressure in the pipe is 3,000 psig after the occurrence of a LBLOCA with minimum safeguards. The maximum allowable pressure under FSAR criteria is determined to be 4,173 psia. There is an approximate 39 percent margin between the calculated maximum pressure and the maximum allowable pressure. Based on this margin, with consideration for the factors discussed above, it is concluded that the pressure in this line under SPU conditions will remain within allowable limits.

Question 3:

In item 49 of the April 12 letter, the licensee indicated that piping systems (i.e., main steam, extract steam, feedwater heater drain and vents, moisture separator and reheater drains, boiler feedwater, and condensate systems) affected by flow increase associated with stretch power uprate, were visually observed to determine if any existing vibration concerns exist. As a follow-up to this visual inspection, walkdowns will be conducted during the increase to SPU power. The acceptance criteria are based on displacement or velocity screening criteria.

Provide a summary of the evaluation for flow effects on the main steam line vibration, which will be increased for the SPU condition. Discuss the plan and schedule of the vibration monitoring program with regard to the power ascension, monitoring methods (installing accelerometers, using hand-held devices), strategic locations of monitoring, and acceptance criteria. Confirm whether the vibration monitoring will be performed for the affected system piping and components in accordance with the American Society of Mechanical Engineers Operations and Maintenance (OM) Code.

Response:

Item 49 of the April 12 letter includes the Indian Point Piping Vibration (PV) Plan Logic to be implemented in support of the SPU. The PV Plan Logic identifies activities to be performed prior to implementation of the uprate and activities to be performed in coordination with the testing program for increasing power to the uprate power level. Activities which have been completed include: (1) Review of PV Condition Reports (CRs) and interviews of key plant personnel regarding PV issues, (2) Documentation of PV acceptance criteria, and (3) Performance of drawing reviews and walkdowns of selected piping systems to identify any existing pre-uprate vibration concerns.

The flow effects on Main Steam Lines are not expected to increase the piping vibration, however vibration monitoring will address any piping vibration not meeting acceptance criteria specified in the piping vibration plan.

Generic Issues and Programs

Vibration monitoring methods and strategic monitoring locations will be addressed in the test plan / procedure developed in support of the PV Plan.

The IP2 PV Plan utilizes the requirements and guidelines of the following reference.

ASME OM-S/G-1994, "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems."

Question 4:

Provide a summary evaluation of the effect of the stretch power uprate on the design basis analysis for high energy line breaks, intermediate energy line breaks, jet impingement and pipe whip restraints.

Response:

Applicable rupture postulation criteria and related design basis documents for Indian Point Unit 2 were reviewed and changes to piping system stress levels resulting from SPU were reconciled against these design basis documents. The evaluations performed concluded that the SPU does not result in any new or revised break locations, and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for SPU.

Question 5:

Section 10.2, "Generic Letter 89-10 Motor-Operated Valve Program," states that the flowrate for the feedwater pump discharge isolation valves will increase due to SPU conditions at IP2. Discuss the evaluation of the increased flowrate on the performance of these MOVs.

Response:

The impact of the increase in feedwater pump discharge isolation valves flowrate on the maximum thrust required to close the valves was evaluated. It was determined that the maximum required thrust occurs after isolation of flow. Therefore, the required thrust to close the valve would not be affected by the increase in flow through the valves under SPU conditions.

Generic Issues and Programs

Question 6:

Section 10.2 states that the changes in system flows, pressures, and temperatures in the NSSSs resulting from the SPU have been documented, and that there are no changes that affect the conclusions of the MOV Program for the NSSS MOVs.

Discuss the changes in system flows, pressure, and temperatures, and the evaluation of the impact on the performance of those MOVs.

Response:

The following is a discussion of changes in flows, pressures, and temperatures resulting from the SPU for the IP2 Nuclear Steam Supply System (NSSS) fluid systems, and the impact of any changes on the conclusions of the MOV Program for the GL 89-10 MOVs in these systems.

- Reactor Coolant System (RCS)

As discussed in Section 4.1.1, the revised parameters that affect RCS performance are core power and the resulting full-load T_{cold} and T_{hot} temperatures. The RCS operating pressure is not changing. Based on the SPU RCS parameters, the RCS design temperature and pressure continue to bound the SPU operating conditions. The RCS System MOV calculations use the RCS design pressure and temperature. Therefore, the SPU does not affect the conclusions of the MOV Program for MOVs in the RCS.

- Chemical & Volume Control System (CVCS)

As noted above, the RCS operating pressure is not changing. With respect to RCS/CVCS interfaces, temperature changes are as noted above for the RCS (Section 4.1.2). Changes in CVCS flow (relative to the slight temperature changes) are considered to be negligible. Based on changes in system parameters being slight / negligible, the SPU does not affect the conclusions of the MOV Program for MOVs in the CVCS.

- Primary Sampling System (PSS)

As noted above, the RCS operating pressure is not changing. No system flow changes are expected. With respect to RCS/CVCS interfaces with the PSS, temperature changes are as noted above for the RCS (Section 4.1.5). The PSS System MOV calculations use the RCS design pressure and temperature, which bound the SPU operating conditions. Therefore, the SPU does not affect the conclusions of the MOV Program for MOVs in the Primary Sampling System.

- Residual Heat Removal (RHR) System

There are no changes in the RHR System operating pressures, flows, and temperatures under SPU conditions. Therefore, the SPU does not affect the conclusions of the MOV Program for MOVs in the RHR System.

Generic Issues and Programs

- Component Cooling Water (CCW) System

There are no changes in the CCW System operating pressures or flows under SPU conditions during normal plant or post-accident operation. Normal operating temperature limits have not changed. Post-LOCA containment sump temperature under SPU conditions remains bounded by the CCW post-LOCA performance Analysis of Record (Section 4.1.6). Accordingly, the SPU does not affect the conclusions of the MOV Program for MOVs in the CCW System.

- High Head Safety Injection (HHSI) System

The HHSI System pump performance is not affected by the SPU. The RWST maximum temperature increases from 100°F to 110°F under SPU conditions (Section 4.1.4). However, RWST temperature changes do not affect the system MOV calculations. Since the system MOV calculations use HHSI pump shutoff head as an input, the SPU does not affect the conclusions of the MOV Program for MOVs in the HHSI System.

- Low Head Safety Injection (LHSI) System

Neither the maximum LHSI flow limits nor pump head performance have changed under SPU conditions. The RWST maximum temperature increases from 100°F to 110°F under SPU conditions (Section 4.1.4). However, RWST temperature changes do not affect the system MOV calculations. Accordingly, the SPU does not affect the conclusions of the MOV Program for MOVs in the LHSI System.

- Recirculation System

Recirculation System pump performance is not affected by the SPU. Since the system MOV calculations use recirculation pump shutoff head as an input, the SPU does not affect the conclusions of the MOV Program for MOVs in the Recirculation System.

- Containment Spray System (CSS)

There are no changes in maximum CSS flow limits under SPU conditions. The RWST maximum temperature increases from 100°F to 110°F under SPU conditions (Section 4.1.4). However, RWST temperature changes do not affect the system MOV calculations. Accordingly, the SPU does not affect the conclusions of the MOV Program for MOVs in the CSS.

<u>Question 7:</u>

Section 10.2 states that the effect of MOV operating parameter changes on related GL 89-10 parameters (e.g., valve dynamic thrust values) has been evaluated and determined to be acceptable.

Discuss the MOV operating parameter changes, the related GL 89-10 parameters, and the evaluation that found those changes to be acceptable.

Generic Issues and Programs

Response:

The impact of the increase in feedwater pump discharge isolation valves flowrate on the maximum thrust required to close the valves is addressed in the response to Generic Issues and Programs, Question 5.

The SPU has no impact on the MOV operating parameters in the Service Water System (Section 9.6).

Based on the response to Generic Issues and Programs, Question 6, there are no changes in operating parameters for MOVs in the NSSS fluid systems resulting from the SPU which affect the valve thrust / torque calculations.

Question 8:

Section 10.2 states that the environmental data review determined that the changes in maximum ambient temperatures at MOV locations are acceptable.

Discuss the maximum ambient temperature changes, and the evaluation that determined the impact on MOV performance to be acceptable (including consideration of Limitorque Technical Update 93-03, as applicable).

Response:

The IP2 evaluation of the effects of reduced motor output torque due to elevated ambient temperatures for MOVs included in the GL 89-10 Program used the following maximum ambient temperatures:

- For MOVs located inside containment, the peak temperature from the pre-uprate accident temperature profile documented in the EQ Program (287°F) was used.
- For MOVs located outside containment, a temperature of 105°F or greater was used.

Evaluation of the impact of the SPU on the above follows:

There is a small increase in the peak containment temperature due to a LOCA under SPU conditions. However, as indicated in Section 10.8.2.2 of the LAR, the pre-uprate accident temperature profile documented in the EQ Program bounds the LOCA temperature profile under SPU conditions.

As indicated in Section 10.8.3.1 of the LAR, normal operating temperatures outside containment remain unchanged under SPU conditions and are bounded by the qualification basis of 105°F.

Accordingly, the SPU does not affect the conclusions of the evaluation of motor torque degradation due to elevated ambient temperature for MOVs inside and outside Containment.

Generic Issues and Programs

Question 9:

Section 10.2 states the analysis of a steamline break inside containment under SPU conditions takes credit for operation of the feedwater control valve isolation MOVs, and that these MOVs will be added to the GL 89-10 program.

Provide the analysis that verifies the capability of the feedwater control valve isolation MOVs to perform their credited function under design-basis conditions (including procurement and maintenance history, actuator sizing and setup calculations, and static and dynamic diagnostic test results).

Response:

The feedwater control valve isolation MOVs (BFD-5 valves) are classified as non-safety related MOVs and were procured as such (Reference 1 allows credit for the use of a backup non-safety grade component (BFD-5 valves) to mitigate the consequences of a postulated pipe break in seismically qualified portions of the main steam line under conditions for which failure of an active component is postulated).

Calculations (Performance Prediction Methodology [PPM], Thrust/Torque including actuator sizing) to support an evaluation of inclusion of the BFD-5 valves into the GL-89-10 program have been performed. Formal documentation for the GL-89-10 program file will be provided prior to operation at the SPU.

Currently, there are auto-generated repetitive PM tasks for Minor PMs and Major PMs for these MOVs. Minor PMs were performed at the end of 2002 on all the feedwater control valve isolation MOVs. The Minor PMs and Major PMs are performed every 2 years and 6 years, respectively.

There are also auto-generated repetitive tasks for static diagnostic testing of these MOVs. All the feedwater control valve isolation MOVs were diagnostically tested in 1995. The static diagnostic test frequency is dictated by the valves' previous as-left margin and risk ranking. Dynamic diagnostic testing on these valves is not practical. However, EPRI PPM calculations will be performed for set-up calculations.

Based on the actions discussed above, the feedwater control valve isolation MOVs are capable of performing isolation of feedwater flow for postulated transients and accidents for which their closure is credited.

Reference:

1. NUREG-0800, Standard Review Plan 15.1.5, "Steam System Piping Failures Inside and Outside Containment (PWR)."

Generic Issues and Programs

Question 10:

Section 10.7, "In-Service Inspection/In-Service Testing Programs," states that the effect of changes on these programs from the SPU will be evaluated as part of the engineering change process.

Discuss, with examples, the evaluation of the impact of the SPU conditions on the performance of safety-related pumps, POVs (including air-operated valves), check valves, and safety or relief valves. Discuss any resulting adjustments to the in-service testing program.

Response:

The following response addresses impact of the SPU on the performance of safety-related pumps and valves in Nuclear Steam Supply System (NSSS) fluid systems and applicable Balance of Plant (BOP) systems, including impact on testing under the IST Program:

NSSS Fluid Systems:

As addressed in the response to Generic Issues and Programs, Question 6, the SPU has either no or negligible impact on system operating pressures and flowrates for the following systems: Reactor Coolant System, Chemical & Volume Control System, Primary Sampling System, Residual Heat Removal System, Component Cooling Water System, Safety Injection System, Recirculation System, and Containment Spray System. As indicated in Section 5.8, the IP2 NSSS pumps and valves are acceptable for the SPU conditions, since the SPU NSSS parameters are bounded by the original NSSS design parameters. Accordingly, the SPU has no impact on IST Program tests for NSSS safety-related pumps and valves.

BOP Systems:

Auxiliary Feedwater (AFW) System

No changes are being implemented for the pumps and valves in the AFW System (Section 9.12). Accordingly, the SPU has no impact on IST Program tests for the safety-related AFW System pumps and valves.

Service Water (SW) System

Increased heat loads due to the SPU result in additional SW flow requirements for some system components. However, the increased flow requirements are within the SW System capability (Section 9.6 of the LAR). The SW pumps are tested at a flowrate of 1500 gpm; the SPU does not affect this test. The SPU does not affect the SW System safety-related MOV operating parameters (e.g., differential pressure), and therefore will not affect IST Program tests for these valves.

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Ventilation (HVAC) Systems

As addressed in Section 9.11.4, the Containment Purge and Pressure Relief System will continue to meet system functional requirements under SPU conditions. No modifications to this system are required to support the SPU. Accordingly, the SPU has no impact on IST Program tests for safety-related valves in this system.

Main Steam (MS) System

Main Steam System safety-related valves include the MSIVs, Non-Return Valves, Main Steam Safety Valves (MSSVs), Atmospheric Steam Relief Valves, and the Turbine Driven Auxiliary Feedwater (AFW) Pump Steam Control Valve. As addressed in Section 9.1.5:

- The MSIVs and Non-Return Valves are of a check valve design, reverse-mounted in series on the MS headers. Reverse flow will assist in closing these valves. Accordingly, under SPU conditions of increased flow, the valves will continue to meet their design capability of closing in 5 seconds or less.

- Maximum steam flow rate at 100 percent power under SPU conditions is significantly below the MSSV capacity. Also, MSSV setpoints are acceptable for operation under SPU conditions.

- Neither the capacity nor the setpoints of the Atmospheric Steam Relief Valves are changing under SPU conditions.

- The Turbine Driven AFW Pump Steam Control Valve reduces the supply pressure to the Turbine Driven AFW Pump to 600 psig or less when MS line pressure is higher than 600 psig. Under SPU conditions at full load, the pressure of the MS supply upstream of the control valve remains above 600 psig, thus providing sufficient pressure for normal operation of the valve.

Based on the above evaluations, the SPU has no impact on IST Program tests for these MS System valves.

Main Feedwater (FW) System

Main Feedwater System safety-related valves include the Feedwater Pump Discharge Isolation Valves, the Main Feedwater Regulating Valves, and the SG Feedwater Inlet Check Valves.

As addressed in Section 9.4 of the LAR, the only Feedwater Pump Discharge Isolation Valve operating parameter affected by the SPU is flowrate. The impact of the increased FW flowrate on the maximum thrust required to close these valves is addressed in the response to Generic Issues and Programs RAI 5. The small increase in flowrate under SPU conditions will not affect the valve closure time.

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The Main Feedwater Regulating Valves will continue to operate within the acceptable control range under SPU conditions.

The SG Feedwater Inlet Check Valves function to prevent leakage of water from the AFW System into the FW System. This function is not affected by the SPU, and therefore the SPU has no impact on IST Program tests for these valves.

Steam Generator Blowdown (SGBD) System

Safety-related Steam Generator Blowdown System valves include the SGBD Containment Isolation Valves. As addressed in Section 9.5, since Feedwater flow increases under SPU conditions, there is an increase in the normal operating continuous blowdown flow under SPU conditions. However, the SPU does not affect the established maximum flow limits for blowdown flow from the steam generators.

Question 11:

Section 10.8.4, "SPU Equipment Qualification Evaluation," states that accident temperatures outside containment in the steam and feedline penetration area have been reanalyzed and result in higher temperatures, and that all equipment outside containment required for accident response have been justified as qualified.

Discuss the evaluation of any safety-related pumps and valves located in the steam and feedline penetration area, and the impact on their performance from higher temperature due to SPU conditions.

Response:

The equipment types in the steam and feedline penetration area on the EQ list are ASCO solenoid valves, Namco limit switches, Westinghouse, Buchanan and Weidmuller terminal blocks, associated cables manufactured by GE and Rockbestos and GE RTV sealant. There are no EQ pumps in this area. The EQ valves evaluated are the ASCO valves.

The ASCO solenoid valves, as well as the other equipment above, were evaluated using the thermal lag analysis of the parts for a 1.0 square foot MSLB header break (limiting break size), winter building ventilation configuration and 102% SPU power. The results are presented in Figure 1 for the ASCO solenoid valves. The temperature of the ASCO case and the coil are very close. However, the coil is only energized for 20 seconds or less to perform the safety function of tripping the MSIVs, so there is little heat generated within the component. After isolation, the MSIV is locked closed. The solenoid valve has no further safety function after the isolation.

The MSIV is a mechanical valve and is not within the scope of the Electrical Equipment EQ Program.

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As shown on Figure 1, the thermal lag temperature of the ASCO coil and case remain below the qualification test temperature. The qualification testing for the ASCO valves included a pre-test accident soak (continuously energized mode) to assure the ASCO valves reached the test chamber temperature.

The cables that are associated with the ASCO solenoid valves are installed in conduit. These cables were also analyzed taking credit for thermal lag. Figure 2 shows the predictions for area temperature and thermal lag as compared to the qualification test profiles. The figure demonstrates that the cables are qualified for the peak temperature and also the long term accident profile.

Therefore, the EQ impact to the safety-related valve equipment from the SPU HELB temperature conditions in the steam and feedline penetration area remain within the test temperatures for which the equipment is demonstrated to be able to perform its intended safety functions.

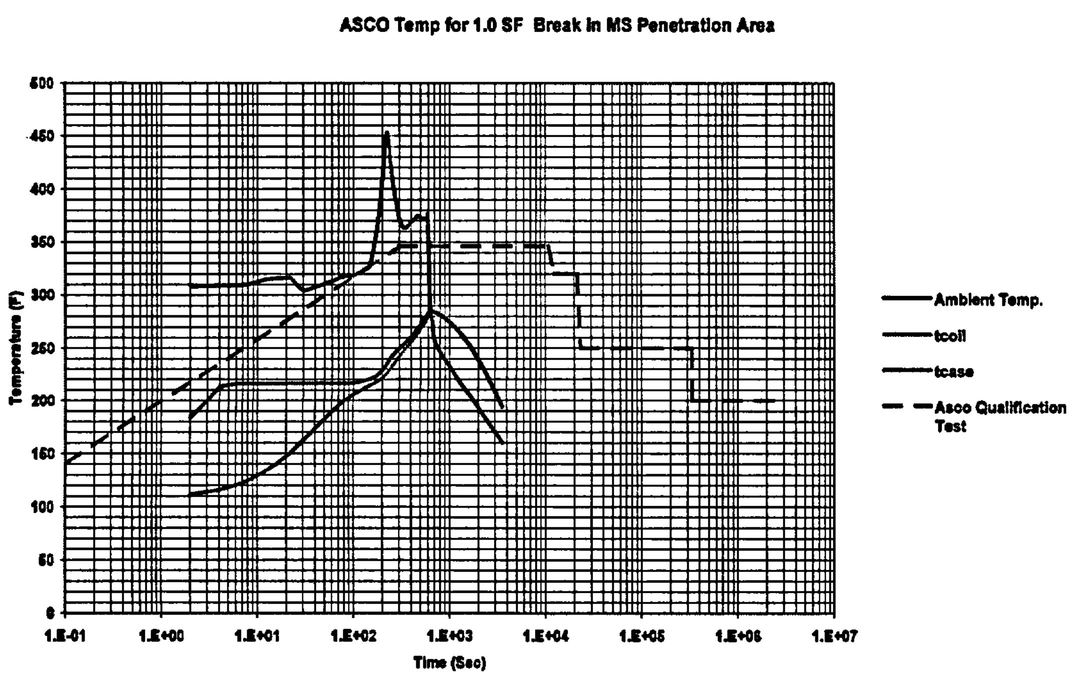

Figure 1: ASCO area, thermal lag and test temperatures

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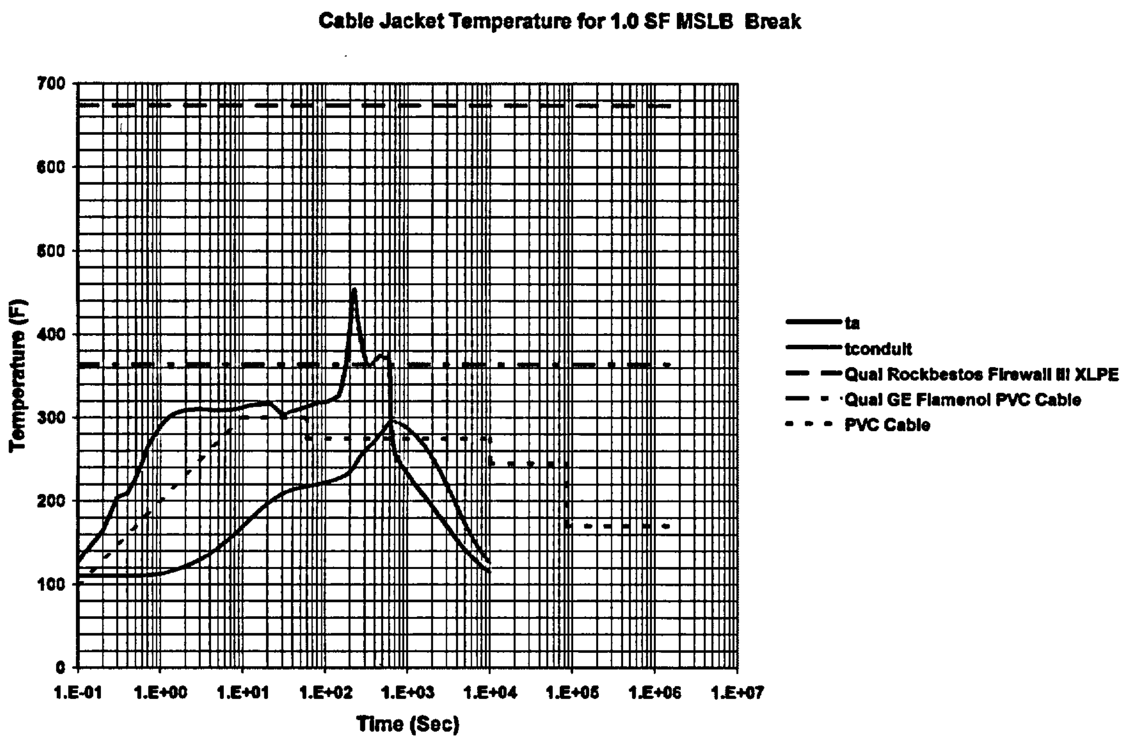

Figure 2: Cable area, thermal lag and test temperatures

Question 12:

Section 10.10, "Generic Letter 95-07," states that the effect of the SPU on the current pressure locking and thermal binding evaluation was reviewed, and that the SPU does not introduce any increased challenge for thermal binding and/or pressure locking and does not effect the results and conclusions of the current evaluation.

Discuss, with examples, the evaluation of the effect of the SPU on the potential for thermal binding and pressure locking of safety-related POVs, including consideration of increased ambient temperatures in applicable locations.

Response:

The results of the IP2 screening of gate valves subject to pressure locking or thermal binding showed the following:

- Seventeen MOVs and two AOVs having double disc parallel seat design were determined to be susceptible to pressure locking.

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- Twelve MOVs having double disc split wedge design were determined to be susceptible to pressure locking.
- Two MOVs having solid wedge design were determined to be susceptible to thermal binding.
- Two MOVs having flexible wedge design were determined to be susceptible to thermal binding.

The following is a summary of the current evaluations and impact of the SPU on these evaluations for MOVs / AOVs subject to pressure locking (PL):

- Seven MOVs (RHR HX Outlet Isolation Valves, CB Spray Header Supply from RHR HX valves, Recirculation Pump Discharge Valves, Containment Sump to RHR Suction Isolation Valve) have a pressure equalization hole drilled in the high pressure side of the valve double disc to eliminate the potential for PL. The SPU does not impact this valve design feature.

- For 4 MOVs (Containment Sump to RHR Suction Isolation Valves, SI Pump Suction from RHR HX Valves), administrative controls (e.g., monitoring pressure in the RHR discharge line to ensure that RCS leakage does not pressurize the RHR System outside of containment) are implemented to prevent PL from occurring. The SPU does not impact use of these administrative controls. In addition, MOV motor up-sizing was performed on these valves.

- For 2 MOVs located outside containment (Emergency Boration Valve, VCT Outlet Valve), pressure induced PL is not a concern because, although the bonnet may pressurize, there is no depressurization of the downstream piping. The SPU does not affect these conditions.

  Thermal induced PL is not a concern because there is no significant change in ambient thermal conditions when the valve is required to function. As indicated in Section 10.8.3.1 of the LAR, normal operating temperatures outside containment remain unchanged under SPU conditions.

- For 2 MOVs (RHR Pump Suction Isolation Valves), it was determined that the valves are not required to function during any DBA, nor are they required to open to achieve safe shutdown. If there is a need to open valves following a non-LOCA transient, the primary plant will be maintained above minimum conditions needed for opening the valves. Accordingly, analysis for PL is not required. The SPU does not affect this evaluation.

- For 4 MOVs located outside containment (CS Pump Discharge Stop Valves), pressure induced PL is not a concern because, when the valve is called upon to open, the upstream pressure from the operating pump will cause the equalization of pressure between the valve upstream pressure and valve bonnet cavity pressure, allowing the valve to open. Also, for these MOVs, administrative controls require that the valves be stroked open and closed following a pump test, thus relieving any pressure that may be trapped in the bonnet. The SPU does not affect these conditions / use of administrative controls.

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Thermal induced PL is not a concern because there is no significant change in ambient temperatures when these valves are required to function. As indicated in Section 10.8.3.1 of the LAR, normal operating temperatures outside containment remain unchanged under SPU conditions.

- Ten MOVs (RHR HX Inlet Isolation Valves, SI Pump Discharge Valves, SI Pump Discharge to SI Injection Header Isolation Valves, CS Pump Discharge to Spray Header Isolation Valves, SI Pump Suction Isolation Valves) and 2 AOVs (Turbine-Driven AFW Pump Steam Isolation Valves) are normally open and are required to be closed for quarterly surveillance testing. These valves are subject to PL when called upon to re-open following the test. However, these valves are eliminated from further evaluation on the basis that they are located in a system / train that is addressed in the Technical Specifications (TSs). Stroking the valve to the closed position renders the associated pump / heat exchanger inoperable, which requires that the plant enter a Limiting Condition of Operation (LCO) per the TSs. The TSs require that the plant be placed a Hot Shutdown condition if the system is not restored to operable status after a defined time limit. The SPU does not affect this evaluation.

  For the 2 AOVs, if the valves were closed, either intentionally or inadvertently, plant procedures direct that a bypass valve be opened to equalize pressure across the valve prior to opening.

- The RHR Pump Discharge Isolation Valve, located outside of containment in the Pipe Penetration Area, was analyzed for pressure locking forces under the following scenario:

  > Following a LOCA, recirculation pump discharge pressure leaks into the valve bonnet and pressurizes the bonnet. Concurrent with the event, there is a loss of ventilation to the area, which is restored within 30 minutes. The ambient temperature in the Pipe Penetration Area increases, resulting in a 1°F increase in the temperature of the trapped bonnet fluid. This temperature increase causes an incremental increase in the bonnet pressure above the recirculation pump discharge pressure. The valve is opened prior to RHR pump start, with upstream and downstream pressures equal to Containment pressure.

  The current analysis shows that the available actuator thrust is greater than the thrust required to open the valve against the maximum bonnet pressure due to pressure locking.

  The SPU does not affect the discharge pressure of the recirculation pumps. Conservatively assuming that the temperature of the trapped bonnet fluid increases by an additional 1°F with resulting increase in bonnet pressure, the thrust required to open the valve against the maximum bonnet pressure is bounded by the MOV's current thrust capability.

The following is a summary of the current evaluations and impact of the SPU on these evaluations for MOVs subject to thermal binding (TB):

- The potential for TB of the Pressurizer PORV Block Valves is addressed, as follows:

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1) The valves are normally closed and may need to open to provide a vent path from the pressurizer to the PRT to mitigate an RCS pressure transient. The valves will not be subject to TB during this event since they would be at a steady state temperature. The SPU does not affect these conditions.

2) The valves are required to be cycled to the open position to arm the Overpressure Protection System when the plant is cooled down. There never has been an event at IP2 where these valves have thermally bound. Therefore, TB due to cool down of the plant is not a concern for these valves.

3) The valves have a type SB actuator, which allows for valve stem growth via a compensating spring pack. Therefore, TB due to valve stem growth is not a concern. The SPU does not impact this valve design feature.

- The potential for TB of the RHR HX CCW Outlet Isolation Valves is addressed, as follows:

1) Under normal operating conditions, prior to being opened, the valve/disc and process fluid temperature will have been in equilibrium near bulk Containment temperature. From plant experience, these valves have exhibited no evidence of TB when opened under these conditions. The SPU does not affect these conditions.

2) In the event of an accident (LOCA / MSLB), these valves receive an immediate signal to open (SI signal). At this point in time, the valve will be operating under parameters that are identical to the normal operating condition, for which there has been no experience of TB. For the LOCA / MSLB event, the containment temperature would rise. However, the valve will either see no temperature change before opening, or if the temperature did increase, the body would be heating up and expanding (rather than contracting) with respect to the disc. Accordingly, TB would not be a concern during these events. The SPU does not affect this evaluation.

Question 13:

Section 10.15.4, "Startup Testing," states that power escalation will be controlled by a specific procedure that includes controls for power escalation, hold points, and data collection requirements. Section 10.15.4 also states that a vibration monitoring activity will be initiated to monitor plant response at various power levels.

Discuss the plans for power escalation including specific hold points and duration, inspections, and plant walkdowns. Also, discuss the vibration monitoring activity including data collection methods and locations, baseline vibration measurements, and planned data evaluation.

Response:

The requested information has been provided in the Entergy letter of April 12, 2004 (NL-04-039) "Supporting Information for License Amendment Request Regarding Indian Point 2 Stretch Power Uprate (TAC MC 1865)". See Attachment II to NL-04-039 response to Question 48 at

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page 25 of 31, and Question 49 at page 30 of 31. Table 2 (page 29) of that response shows planned data collection at various hold points.

Question 14:

Discuss the evaluation of potential flow vibration effects resulting from SPU conditions for reactor pressure vessel internals, and steam and feedwater systems and their associated components, including impact on structural capability and performance during normal operations, anticipated transients (initiation and response), and design-basis conditions; and preparation for responding to the potential occurrence of loose parts as a result of the power uprate.

Response:

- Reactor Vessel Internals

Flow induced vibrations (FIV) of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies is to assure the structural integrity and reliability of the reactor internals components. These efforts have included in-plant tests, scale model tests, tests in fabricators' shops, bench tests of components, and various analytical investigations. The results of scale model and in-plant tests indicate that the vibrational behavior of 2-, 3-, and 4-loop plants is essentially similar; the results obtained from each of the tests complement one another and make possible a better understanding of the flow induced vibration phenomena.

As described in References 1 and 2, Westinghouse performed a comprehensive instrumented reactor internals testing program at the Indian Point Unit 2 plant. This test program included heatup and cooldown as well as operation with 1, 2, 3, and 4 reactor coolant pumps, including starting and stopping transient operations. The initial program was performed without the core present (Reference 1). A subsequent program was performed with the core in place (Reference 2). The results of this program were used to develop theories and concepts related to reactor internals vibration under various operating conditions as well as to assess the fatigue and stress effects of operational vibrations. The testing performed at Indian Point 2 included the acquisition of data during hot functional testing (without core present) and subsequently with the core installed. The results of this comprehensive testing program showed that the vibrational response of the reactor internals is small and that adequate margins of safety exist with regard to flow induced vibration.

To address the SPU program at IP2 an evaluation was performed to show that the vibration characteristics of reactor internals do not change significantly and the structural adequacy of the reactor internals in regards to FIV is not impaired.

The reactor internal components that are generally addressed for FIV consists of lower internals (core barrel, thermal shield support flexures, thermal shield support bolts and dowel pins) and upper internals (guide tubes). The current design temperature range between T_{cold} and T_{hot} is

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67.2°F for the low T_{avg} range and 65.0°F for the high T_{avg} range and changes to 69.4°F for the low T_{avg} range and 67.6°F for the high T_{avg} range with the implementation of SPU at IP2.

This SPU design condition will slightly alter T_{cold} and T_{hot} fluid densities, which will slightly change the forces, induced by flow. The corresponding T_{cold} and T_{hot} fluid densities will increase by about 2%.

Evaluations performed for the SPU conditions show that the FIV loads on the guide tubes and the upper support columns increases by about 9% and the impact on the lower internals is negligible. Benchmark tests of guide tubes and upper support columns together with previous FIV analysis for similar 4-loop reactors has shown that a large margin exists in regards to calculated stresses versus the code allowable. Therefore, the effect on the FIV on the reactor internals is considered negligible or essentially non-existent for the SPU conditions at the IP2 plant.

References:

1. WCAP-7879-P-A, "Four Loop PWR Internals Assurance and Test Program", July 1972.
2. WCAP-7879-AD1, "Four Loop PWR Internals Assurance and Test Program Addendum 1, IPP-2 Reactor Internals Vibration with-Core Testing Program", October 1972.

- Steam Generator

Steam generator tube vibration and wear are addressed in Section 5.6.6 of the LAR.

<u>Steam and Feedwater Systems and Their Associated Components</u>

The main steam and feedwater piping systems and their associated components will be evaluated for potential flow vibration effects resulting from SPU conditions. These piping systems will be included in the piping vibration monitoring plan to be performed in support of SPU. The piping vibration monitoring plan will identify the specific piping locations for monitoring, the monitoring methods to be used (e.g., accelerometers, hand held devices), as well as acceptance criteria to determine piping vibration acceptability.

Refer to response for Generic Issues and Programs Question 3 for additional details related to the overall piping vibration monitoring plan.

<u>Response to the potential occurrence of loose parts as a result of the power uprate.</u>

Entergy has procedures in place for the control of and exclusion of foreign objects during maintenance activities, including during outages. These procedures have been successful in controlling foreign objects. Entergy has installed metal impact monitors to detect the occurrence of loose parts or foreign objects in the reactor coolant system. Detection of unusual signals from the metal impact monitors triggers investigations and evaluations to determine the source of the signals and to take corrective actions if that is needed.