Dominion Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, Virginia 23060
Web Address: www.dom.com

Dominion®

January 14, 2008

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2378

Serial No.: 07-0834D
NLOS/MAE: R0
Docket No.: 50-423
License No.: NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST
RESPONSE TO QUESTIONS EMCB-07-0060 THROUGH EMCB-07-0069 AND EMCB-07-0071 THROUGH EMCB-07-0081

Dominion Nuclear Connecticut, Inc. (DNC) submitted a stretch power uprate license amendment request (LAR) for Millstone Power Station Unit 3 (MPS3) in letters dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A), and supplemented the submittal by letters dated September 12, 2007 (Serial No. 07-0450B) and December 13, 2007 (Serial No. 07-0450C). The NRC staff forwarded requests for additional information (RAIs) in October 29, 2007 and November 26, 2007 letters. DNC responded to the RAIs in letters dated November 19, 2007 (Serial No. 07-0751) and December 17, 2007 (Serial No. 07-0799). The NRC staff forwarded an additional RAI in a December 14, 2007 letter. The response to questions EMCB-07-0060 through EMCB-07-0069 and questions EMCB-07-0071 through EMCB-07-0081 of this RAI is provided in the attachment to this letter.

The information provided by this letter does not affect the conclusions of the significant hazards consideration discussion in the December 13, 2007 DNC letter (Serial No. 07-0450C).

Should you have any questions in regard to this submittal, please contact Ms. Margaret Earle at 804-273-2768.

Sincerely,

Gerald T. Bischof
Vice President Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Gerald T. Bischof, who is Vice President Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 14th day of January, 2008.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

MARGARET B. BENNETT
Notary Public 354302
Commonwealth of Virginia
My Commission Expires Aug 31, 2008

Commitments made in this letter: None

Attachment

cc: U.S. Nuclear Regulatory Commission
Region I
Regional Administrator
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. J. G. Lamb
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop O-8B1A
Rockville, MD 20852-2738

Ms. C. J. Sanders
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop O-8B3
Rockville, MD 20852-2738

Mr. S. W. Shaffer
NRC Senior Resident Inspector
Millstone Power Station

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

ATTACHMENT

LICENSE AMENDMENT REQUEST

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RESPONSE TO QUESTIONS EMCB-07-0060 THROUGH EMCB-07-0069 and EMCB-07-0071 THROUGH EMCB-07-0081

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

Mechanical and Civil Engineering Branch

EMCB-07-0060

The postulated pipe break acceptance criteria inside and outside containment are described in FSAR Sections 3.6.1 and 3.6.2 and reflect the approach and methodology contained in the Branch Technical Positions ASB 3-1 and MEB 3-1. SPULAR Attachment 5 Section 2.2.1.2.2 states that "The SPU evaluations performed for applicable piping systems did not result in any new or revised break/crack locations, and the design basis for pipe break, jet impingement, pipe whip and environmental considerations remain valid for SPU" and "Pipe stresses for break exclusion zones were demonstrated to be within acceptable limits".

a) Confirm whether these analyses included reactor coolant loop (RCL) branch line break, pressurizer surge line break, main SLB and feedwater line break. If not, provide technical justification for not including these pipe breaks.
b) Provide a summary description of the evaluations, explaining how the evaluations were performed. Include assumptions and load combinations along with summaries of results that show that you meet the FSAR pipe break acceptance criteria when SPU conditions are included.

DNC Response

a) The analyses included the RCL branch line breaks including the pressurizer surge line break. Main steam and feedwater line breaks were also included in the analysis.

b) For piping systems where pipe stress levels were impacted by SPU, the applicable load combinations were used in accordance with the existing Millstone 3 licensing and design basis including FSAR Sections 3.6.1 and 3.6.2 and conforms to BTP ASB 3-1 and MEB 3-1. All affected piping systems had no new locations where the total additive pipe stress level exceeded the allowable limit of .8 (1.2 Sh + Sa).

EMCB-07-0061

Section 2.2.2.1.2.2 states that "By virtue of LBB [leak before break], breaks are not postulated for the RCL loop hot leg, cold leg and crossover leg piping".

a) Confirm whether the current licensing basis is based on LBB methodology.
b) Provide justification that the basis for using LBB methodology is still valid under the proposed SPU conditions.

DNC Response

a) DNC confirms that the current licensing basis for MPS3 is based on the primary loop piping LBB methodology.

b) Justification that the basis for using LBB methodology is still valid under the proposed SPU conditions is given in the SPU licensing report Section 2.1.6.

EMCB-07-0062

Section 2.2.2.1.2.2 states that "For the SPU program, the loop LOCA [loss-of-coolant accident] hydraulic forcing function forces and associated loop LOCA RPV [reactor pressure vessel] motions from applicable RCL branch line breaks were reconciled as part of the RCL and associated branch piping and support evaluations." Identify RCL branch line breaks used for loop LOCA analysis and describe the method used for reconciliation.

DNC Response

The RCL branch line breaks that were considered included the Residual Heat Removal pump suction lines off of the Loop A and Loop D hot legs; the Pressurizer Surge line off of the Loop B hot leg; and the Safety Injection line off of Loops A, B, C and D. Dynamic forcing functions were developed for these seven branch line breaks and included consideration of broken loop and unbroken loop applications. The dynamic forcing functions were input into a piping structural model and piping stresses, pipe support loads and primary equipment support loads were developed and reconciled to existing design basis criteria.

EMCB-07-0063

Section 2.2.2.1.2.3 indicates that the stress results shown in Table 2.2.2.1-1 have incorporated the hydraulic LOCA forces.

a) Provide the basis for the allowable values and the loading combinations used for the calculated stresses in Table 2.2.2.1-1.
b) Footnote 5 of the table states that the allowable levels are well below material yield. Provide the corresponding yield values that confirm this statement.
c) Provide the basis for the allowable stress of 25,050 pounds per square inch (psi) for the 10" safety injection cold leg Loop D line.
d) For the cumulative usage factor (CUF) values that exceed 0.1, verify that these locations are postulated pipe breaks.
e) Also confirm whether the CUF values in the "SPU" and "Current" columns are to the end of the 60 year plant life in accordance with the licensing renewal of the plant.

DNC Response

a) The basis for the allowable values and the loading combinations is the existing Millstone Unit 3 licensing and design basis which is in compliance with ASME Section III and ANSI B31.1.

b) The end of this sentence should have read that the allowable stress levels are well below material ultimate stress limits, instead of stating that allowable stress levels are well below material yield and/or ultimate stress levels.

c) The allowable stress of 25,050 should have read 50,100 psi, based on an allowable stress value equal to 3.0 Sm, where Sm = 16,700 psi per ASME III Code applicable to MP3, for stainless steel material SA-376 TP-316.

d) All locations where the CUF exceeds 0.1 are postulated pipe break locations. These locations are postulated pipe break locations in the current design basis. There are no new locations as a result of SPU where the CUF exceeds 0.1.

e) The "SPU" column in Table 2.2.2.1-1 represents CUF based upon a 60-year plant life. The "Current" column CUF numeric values come from pre-license renewal engineering analysis (i.e., 40-year plant life). However, the "Current" column CUF can be considered as representing a 60-year plant life as well.

   The MPS3 license renewal submittal contained the following statement:

   "The fatigue design bases (e.g., design transients) for MPS3 has been reviewed and found to be acceptable for the 60-year license renewal period."

   Therefore, there was no change in the design transient number of cycles due to license renewal. Thus, the "Current" column represents CUF's based upon pre-SPU process conditions and a 60-year plant life (i.e., the post-license renewal number of transient cycles).

EMCB-07-0064

Section 2.2.2.2.2.2 states that "The BOP [balance of plant] piping and support systems listed in Section 2.2.2.2.2.1 (Introduction) have been evaluated relative to the impact of SPU." Thermal, pressure and flow change factors equal to the ratio of SPU to actual analyzed value were determined. "For change factors greater than 1.00, an additional evaluation was performed to address the specific increase in temperature, pressure and/or flow rate in order to determine piping and support system acceptability, as well as nozzle load and containment penetration acceptability."

a) List all systems (inside and outside containment) with "change factors" greater than 1.00.

b) For systems with “change factors” greater than 1.00, provide the method of your evaluation. Provide a quantitative summary of the maximum stresses and fatigue usage factors (if applicable) for original and SPU conditions with a comparison to code of record allowable stresses. Include only maximum stresses and data at critical locations (i.e., nozzles, penetrations, etc). List all pipe system modifications (for pipe supports see (d) below) required due to SPU and schedule of completion. For affected nozzles and containment penetrations, provide a summary of loads compared to specific allowable values for nozzles and penetrations.

c) For systems with a thermal change factor greater than 1.00, provide a description of preoperational measures taken to ensure that thermal expansion will not impose an unanalyzed condition that could potentially overstress piping and supports. In addition, confirm that a program will be in place for monitoring thermal expansion at the startup of the SPU.

d) For systems in (b), state the method used for evaluating pipe supports when considering SPU conditions and confirm that the supports on affected piping systems will remain structurally adequate to perform their intended design function. Provide detail descriptions of all pipe support modifications needed to meet design basis at SPU conditions. Also list type, size, loading (current and SPU) and location of supports that need to be modified and added due to SPU conditions.

e) Provide schedule of completion for all piping and pipe support modifications and additions.

DNC Response

a) Portions of the feedwater, condensate, feedwater heater vents and drains, moisture separator vents and drains, and component cooling water piping systems contained change factors greater than 1.00.

b) For piping systems containing change factors greater than 1.00, these piping systems were evaluated using simplified hand calculation methods (manually increasing existing stresses and loads) or by performing more detailed computer analyses. For example, if a piping temperature increased from 150°F to 160°F due to SPU, the resulting thermal change factor would be equal to 1.13 based on the ratio of (160-70)/(150-70). The existing thermal expansion stress levels and support loads based on 150°F would be increased by 13 percent to determine the corresponding values at 160°F. The revised thermal expansion stress levels would then be demonstrated to be less than the applicable allowable stress limit for this loading condition. The revised thermal expansion pipe support loads would be combined with other concurrent loadings (e.g., deadweight, seismic), to determine a revised pipe support design load. This revised design load would then be demonstrated to be acceptable for the applicable pipe support components. In cases where simplified hand calculation methods were not utilized, more detailed pipe stress and/or pipe support computer analyses were used to demonstrate component acceptability.

A summary of the maximum stress levels for current and SPU conditions including a comparison to code of record allowable stress levels is provided in Table 2.2.2.2-1. For each piping system listed in this table, the stresses reported are at the most critical location of the piping system, corresponding to the piping location containing the highest design margin ratio (i.e., design margin ratio is defined as the ratio of SPU stress divided by the allowable stress). These critical stress locations may be at equipment nozzles, containment penetrations, or any in line piping component (e.g., valve, elbow, or reducer) within the analytical boundaries of the piping stress model.

There were no piping modifications (i.e., physical piping re-routes) required due to SPU. Details related to pipe support modifications are provided in response to Item d) below.

A summary of SPU loads and/or stresses and related allowable values for nozzles and penetrations that were most affected by SPU are as follows.

1st Point Feedwater Heaters 3FWS-E1A, 3FWS-E1B and 3FWS-E1C
Maximum SPU stress = 9552 psi (for 3FWS-E1B) which is less than the allowable stress of 13,125 psi

3rd Point Feedwater Heaters 3CNM-E3A, 3CNM-E3B and 3CNM-E3C
Maximum SPU stress = 1842 psi (for 3CNM-E3C) which is less than the allowable stress of 13,125 psi

4th Point Feedwater Heaters 3CNM-E4A, 3CNM-E4B and 3CNM-E4C
Maximum SPU stress = 11338 psi (for 3CNM-E4B) which is less than the allowable stress of 13,125 psi

6th Point Feedwater Heaters 3CNM-E6A, 3CNM-E6B and 3CNM-E6C
Maximum SPU stress = 3326 psi (for 3CNM-E6A, B and C) which is less than the allowable stress of 13,125 psi

Heater Drain Pump 3DSM-P1A
Governing SPU Loads are for 3DSM-P1A as follows:
Axial Force = 253 lbs which is less than the allowable axial load of 3500 lbs
Maximum Lateral Force = 426 lbs which is less than the maximum lateral force allowable load of 2500 lbs
Torsional Moment = 5545 ft-lbs which is less than the allowable torsional moment of 6000 ft-lbs
Maximum Bending Moment = 4068 ft-lbs which is less than the bending allowable moment of 5000 ft-lbs

Component Cooling Water Pumps 3CCP*P1A, 3CCP*P1B and 3CCP*P1C

3CCP*P1A (Suction Nozzle)
Normal/Upset Stress = 16041 psi which is less than the allowable stress of 21000 psi
Faulted Stress = 16880 psi which is less than the allowable stress of 31500 psi
3CCP*P1A (Discharge Nozzle)
Normal/Upset Stress = 17883 psi which is less than the allowable stress of 21000 psi
Faulted Stress = 20649 psi which is less than the allowable stress of 31500 psi

3CCP*P1B (Suction Nozzle)
Normal/Upset Stress = 13813 psi which is less than the allowable stress of 21000 psi
Faulted Stress = 14643 psi which is less than the allowable stress of 31500 psi
3CCP*P1B (Discharge Nozzle)
Normal/Upset Stress = 13299 psi which is less than the allowable stress of 21000 psi
Faulted Stress = 16185 psi which is less than the allowable stress of 31500 psi

3CCP*P1C (Suction Nozzle)
Normal/Upset Stress = 10149 psi which is less than the allowable stress of 21000 psi
Faulted Stress = 11144 psi which is less than the allowable stress of 31500 psi
3CCP*P1C (Discharge Nozzle)
Normal/Upset Stress = 15614 psi which is less than the allowable stress of 21000 psi
Faulted Stress = 17602 psi which is less than the allowable stress of 31500 psi

Main Feedwater Pumps 3FWS-P1, 3FWS-P2A, 3FWS-P2B

3FWS-P1 (Suction Nozzle)						
Loading Condition	Fx (lbs)	Fy (lbs) (Axial)	Fz (lbs)	Mx (ft-lbs)	My (ft-lbs) (Torsional)	Mz (ft-lbs)
Deadload	428	7150	557	1494	1355	138
Allowable Deadload	500	8000	700	1700	1600	200
Thermal	244	2978	4808	19420	42048	9185
Allowable Thermal	1000	4000	6000	22000	47000	11000

3FWS-P1 (Discharge Nozzle)						
Loading Condition	Fx (lbs)	Fy (lbs)	Fz (lbs) (Axial)	Mx (ft-lbs)	My (ft-lbs)	Mz (ft-lbs) (Torsional)
Deadload	179	2047	43	1823	44	8131
Allowable Deadload	200	2300	100	2000	100	9000
Thermal	13989	455	5367	39541	6348	12689
Allowable Thermal	16000	1000	6000	44000	7000	14000
Fluid Transient	14859	7712	11415	86985	19347	31276
Allowable Fluid Transient	17000	9000	13000	93000	22000	35000

3FWS-P2A (Suction Nozzle)						
Loading Condition	Fx (lbs)	Fy (lbs) (Axial)	Fz (lbs)	Mx (ft-lbs)	My (ft-lbs) (Torsional)	Mz (ft-lbs)
Deadload	120	11644	740	7982	5645	5980
Allowable Deadload	200	13000	900	8900	6300	6700
Thermal	982	979	882	5732	9699	14964
Allowable Thermal	2000	2000	1000	7000	12000	19000
3FWS-P2A (Discharge Nozzle)						
Loading Condition	Fx (lbs)	Fy (lbs)	Fz (lbs) (Axial)	Mx (ft-lbs)	My (ft-lbs)	Mz (ft-lbs) (Torsional)
Deadload	312	6232	1883	7318	206	1414
Allowable Deadload	400	6900	2100	8100	300	1600
Thermal	1023	19301	12769	981	16390	5729
Allowable Thermal	2000	22000	15000	2000	19000	7000
Fluid Transient	4728	5588	35593	6100	38933	24156
Allowable Fluid Transient	6000	7000	40000	7000	43000	28000

3FWS-P2B (Suction Nozzle)						
Loading Condition	Fx (lbs)	Fy (lbs) (Axial)	Fz (lbs)	Mx (ft-lbs)	My (ft-lbs) (Torsional)	Mz (ft-lbs)
Deadload	118	8752	707	1729	3748	1972
Allowable Deadload	200	9800	800	2000	4200	2200
Thermal	1268	386	2271	16964	3798	12017
Allowable Thermal	2000	1000	3000	19000	5000	16000
3FWS-P2B (Discharge Nozzle)						
Loading Condition	Fx (lbs)	Fy (lbs)	Fz (lbs) (Axial)	Mx (ft-lbs)	My (ft-lbs)	Mz (ft-lbs) (Torsional)
Deadload	443	7398	560	7173	23	664
Allowable Deadload	500	8200	700	7900	100	800
Thermal	8055	12916	2871	498	1410	10659
Allowable Thermal	9000	15000	4000	1000	2000	12000
Fluid Transient	6607	2147	16362	2971	3455	14107
Allowable Fluid Transient	8000	3000	18000	4000	4000	16000

Containment Penetrations 5 and 6 (Feedwater Loops A & B)
Normal/Upset Stress = 12261 psi which is less than the allowable stress of 21000 psi

Containment Penetrations 7 and 8 (Feedwater Loops C & D)
Normal/Upset Stress = 11790 psi which is less than the allowable stress of 21000 psi

c) During the baseline walkdown performed for piping vibration, piping systems subjected to a temperature increase associated with SPU were inspected to identify any locations where there was a potential for unacceptable thermal expansion interaction. None were noted. The increases in thermal expansion displacements associated with SPU are less than 1/16 inch and, therefore, these increased displacements are not a concern. However, during startup of the SPU, piping systems will be observed to identify any unanticipated unacceptable conditions.

d) For pipe supports on systems containing change factors greater than 1.00, these pipe supports were evaluated either using simplified hand calculation methods (manually increasing existing loads) or by performing more detailed computer analyses, in order to reconcile the specific support load increases. For example, if a piping temperature increased from 150°F to 160°F due to SPU, the resulting thermal change factor would be equal to 1.13 based on the ratio of (160-70)/(150-70). The existing thermal expansion support loads based on 150°F would be increased by 13 percent to determine the corresponding values at 160°F. These revised thermal expansion pipe support loads would then be combined with other concurrent loadings (e.g., deadweight, seismic), to determine a revised pipe support design load for SPU. Applicable pipe supports would then be evaluated using these revised design loads in order to demonstrate that stresses and loads for affected pipe support components remain within acceptable design basis limits. In cases where simplified hand calculation methods were not utilized, more detailed pipe support computer analyses were used to demonstrate pipe support component acceptability.

Details (e.g., support mark number, location, modification description, current loads, SPU loads) related to pipe supports that needed to be modified due to SPU conditions are as follows.

3-CCP-3-PSR079 (Strut located in Auxiliary Building)
Existing strut assembly being replaced with higher capacity strut assembly
Current Load = 27440 lbs
SPU Load = 30320 lbs

3-CCP-3-PSR464 (Rigid support located in Auxiliary/Fuel Building Tunnel)
Support will be modified by adding additional welds
Current Loads are Fx = 21175 lbs, Fy = 9161 lbs, Fz = 2407 lbs
SPU Loads are Fx = 22937 lbs, Fy = 11337 lbs, Fz = 2493 lbs

3-RSS-1-PSA097 (Anchor located in Containment Building)
Anchor will be modified by stiffening existing members and adding related welds
Current Loads are Fx = 9690 lbs, Fy = 1 lbs, Fz = 27312 lbs
Mx = 2 ft-lbs, My = 33105 ft-lbs, Mz = 22 ft-lbs
SPU Loads are Fx = 14709 lbs, Fy = 1 lbs, Fz = 36098 lbs
Mx = 2 ft-lbs, My = 39932 ft-lbs, Mz = 26 ft-lbs

3-RSS-1-PSA098 (Anchor located in Containment Building)
Anchor will be modified by stiffening existing members and adding related welds
Current Loads are Fx = 9854 lbs, Fy = 2 lbs, Fz = 27886 lbs
Mx = 0 ft-lbs, My = 33933 ft-lbs, Mz = 33 ft-lbs
SPU Loads are Fx = 14890 lbs, Fy = 3 lbs, Fz = 36846 lbs
Mx = 0 ft-lbs, My = 40641 ft-lbs, Mz = 42 ft-lbs

3-FWS-5-PSR144 (Rigid support located in the Turbine Building)

Support will be modified by providing additional weld
Current Loads are Fy = 36674 lbs, Fz = 12186 lbs
SPU Loads are Fy = 41282 lbs, Fz = 15931 lbs

3-FWS-5-PSR135 (Rigid support located in the Turbine Building)
Support will be modified by adding additional tube steel stiffener members and associated welds.
Current Loads are Fx = 38058 lbs, Fy = 26171 lbs
SPU Loads are Fx = 34760 lbs, Fy = 48982 lbs

3-FWS-5-PSR505 (Rigid support located in the Turbine Building)
Support will be modified by adding additional tube steel and stiffener plate members and associated welds
Current Load = 1145 lbs
SPU Load = 2676 lbs

3-FWR-5-PSR006 (Rigid support located in the Turbine Building)
Support will be modified by adding shims and welds
Current Load = 1649 lbs
SPU Load = 5790 lbs

3-FWS-5-PSR-106 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing struts with higher load capacity struts
Current Load = 26416 lbs
SPU Load = 44429 lbs

3-FWS-5-PSR-147 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing strut with higher load capacity strut
Current Load = 8540 lbs
SPU Load = 13121 lbs

3-FWS-5-PSR148 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing strut with higher load capacity strut
Current Loads are Fy = 44236 lbs, Fz = 17669 lbs
SPU Loads are Fy = 48151 lbs, Fz = 22173 lbs

3-FWS-5-PSSH602 (New spring hanger support located in the Turbine Building)
SPU Load = 20000 lbs

3-FWS-5-PSSH603 (New spring hanger support located in the Turbine Building)
SPU Load = 20000 lbs

3-FWS-5-PSR141 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing strut with higher load capacity strut

Current Load = 37475 lbs
SPU Load = 71734 lbs

3-FWS-5-PSSP460 and 461 (Snubbers located in the Turbine Building)
The support frame structure to which these snubbers are attached will be modified
Current Load = 102002 lbs
SPU Load = 172331 lbs

3-FWS-5-PSR134 (Rigid support located in the Turbine Building)
Support will be modified by adding new plate to achieve additional weld footprint
Current Load = 26273 lbs
SPU Load = 43023 lbs

3-FWS-5-PSSP464 (Snubber located in the Turbine Building)
Support will be modified by replacing existing snubber with higher load capacity snubber
Current Load = 11044 lbs
SPU Load = 22681 lbs

3-FWS-5-PSST507 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing spring hanger with strut
Current Load = 847 lbs
SPU Load = 3573 lbs

3-FWS-5-PSR508 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing rod hanger with strut
Current Load = 550 lbs
SPU Load = 1459 lbs

3-FWS-5-PSST600 (New strut located in the Turbine Building)
SPU Load = 3154 lbs

3-FWS-5-PSR601 (New rigid support located in the Turbine Building)
SPU Load = 3762 lbs

3-INF-6-PSRH109 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing clamp with higher load capacity pipe clamp
Current Load = 2000 lbs
SPU Load = 3895 lbs

3-INF-6-PSRH110 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing rods and associated hardware with higher load capacity components

Current Load = 9000 lbs
SPU Load = 12251 lbs

30EFF-6-PSRH157 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing rods and associated hardware with higher load capacity components
Current Load = 2000 lbs
SPU Load = 5377 lbs

3-EFF-6-PSRH158 (Rigid support located in the Turbine Building)
Support will be modified by replacing existing rods and associated hardware with higher load capacity components
Current Load = 2000 lbs
SPU Load = 5246 lbs

e) Piping and support modifications required for SPU will be completed prior to increasing the reactor core power above 3411-megawatts thermal (current license condition). Reactor core power operation above 3411-megawatts is scheduled to occur after the Fall 2008 refueling outage.

EMCB-07-0065

Section 2.2.2.2.2.2 states that "...applicable feedwater system pipe supports were evaluated and demonstrated to be within design basis limits." Section 2.2.2.2.2.3 states that "The piping evaluations also concluded that the feedwater system can withstand water hammer loads associated with SPU conditions (resulting from a feedwater isolation valve closure/pump trip event) although several pipe support modifications will be required." Provide explanation for the apparent discrepancy between these statements.

DNC Response

Section 2.2.2.2.2.3 provides the most complete statement. Approximately seventeen feedwater pipe support modifications are required to accommodate the increased loadings associated with SPU. Section 2.2.2.2.2.2 discusses the feedwater pipe support system which satisfies the piping and support design basis. However, an individual component within the individual pipe support may have its design limit exceeded and require a modification such as increasing a weld size.

EMCB-07-0066

Margin, as defined for values in Table 2.2.2.2-1 (and in Table 2.2.2.1-1 as stated in second line of Section 2.2.2.1.2.3), is not clear. For instance, the example mentioned in Note 2 of Table 2.2.2.2-1 defines margin as the ratio of the calculated value to the allowable value. Typically, margin is defined by the difference between the allowable value and the calculated value divided by the allowable value. Provide justification for the margin definition used for these tables.

DNC Response

The margin definition used in Table 2.2.2.2-1 is the ratio of calculated stress/allowable stress. This definition can be referred to as Design Interaction Ratio. The justification for using this definition is to provide a ratio of less than 1.0.

EMCB-07-0067

Section 2.12.1.2.3.2, Vibration Monitoring, states that "SPU implementation will result in higher flow rates for piping systems within the main power cycle. Secondary system piping and supports evaluated included the following: main steam, extraction steam, feedwater, condensate, feedwater heater vents and drains and moisture separator vents and drains. The evaluations concluded that piping systems remain acceptable and will continue to satisfy design basis requirements. Piping vibration reviews, including system walk-downs, will be performed during power ascension to the SPU level, to ensure piping system and component vibrations remain acceptable." Section 2.2.2.2.2.3 also indicates that these systems will be reviewed for flow induced vibration (FIV) issues at a later time. These statements are confusing since in one place they imply that piping "evaluations" for FIV for the higher SPU flow rate have been completed and in another place it is indicated that piping reviews for FIV will be performed at a later date. In addition, during a telecon between the staff and DNC, DNC indicated that an evaluation for FIV due to higher SPU flow rates on affected BOP systems (see above) will be performed after a collection of vibration data at 100 percent current licensed thermal power (CLTP) to establish a baseline has been completed, which is scheduled to be performed in November 2008. Provide a clear description of the planned activities and evaluation methodology. Also, provide the acceptance criteria for the evaluation of FIV for these piping systems as well as evaluation summaries which show that the acceptance criteria have been met for SPU conditions.

DNC Response

DNC has developed a comprehensive plan to address flow induced vibration in piping affected by the power uprate. The plan began with the development of a program to address scope, method, evaluation and acceptance criteria. The scope includes all piping with increased flow rates resulting from the power uprate. The method was to perform a series of walkdowns spanning from the current plant condition to the completion of power ascension testing following implementation of the power uprate. The pre-baseline walkdowns were performed and completed on October 22-23, 2007. Those walkdowns for the current plant condition identified no current non-conforming conditions. The evaluation of the current plant condition concluded that the piping systems will remain acceptable at SPU conditions and will continue to satisfy design basis requirements. To validate that the piping systems will remain acceptable at SPU conditions, approximately fifty locations were identified where detailed observations will be performed during power uprate implementation. The acceptance criteria for all piping evaluations will be in accordance with ASME OM Part 3.

EMCB-07-0068

a) Identify any pressure retaining systems (besides the ones listed in EMCB- 07-0067) that would experience higher flow rates due to the SPU implementation.

b) Describe the methodology and provide the acceptance criteria for the evaluation of FIV for these systems along with evaluation summaries which show that the acceptance criteria have been met.

DNC Response

a) Systems that will see increased flow rates due to power uprate include main steam, feedwater, condensate, extraction steam, feedwater heater vents and drains, and moisture separator vents and drains.

b) The methodology for evaluation and acceptance criteria for all piping evaluated for vibration issues will be in accordance with ASME OM Part 3. Also see responses to EMCB-07-0067 and EMCB-07-0069.

EMCB-07-0069

Describe the vibration monitoring program at the startup for the SPU implementation, its basis and acceptance criteria. Confirm whether it is in accordance with the ASME OM Code Part 3.

DNC Response

Piping systems that will experience increased flow rates due to SPU will be inspected using visual methods during SPU implementation. Initially simple tools and methods as described in ASME OM Part 3 will be used. If warranted, hand-held instrumentation will be employed to record data. The entire piping vibration plan for Millstone Unit 3 SPU is in accordance with ASME OM Part 3.

EMCB-07-0071

Table 2.2.3-3 states that for the case of the Core Barrel Outlet Nozzle Section A-A, which exceeded the code allowable limit of 3 Sm, the "simplified elastic-plastic analysis was performed to calculate fatigue strength, as allowed by ASME, B&PV Code, Section III, NB 3228.5. These conditions have been met and the fatigue usage is less than 1.0." Provide a summary of the evaluation which shows that the special rules for exceeding

$3S_m$ as provided by (a) through (f) of Subparagraph 3228.5 have been met.

DNC Response

According to Section NG-3228.3, the $3S_m$ limit may be exceeded, provided that the requirements listed in that section are met.

(a) The range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses, shall be less than $3S_m$.

The stress intensity is 25,362 psi < $3S_m$ = 34,440 psi

(b) The value of S_a used for entering the design fatigue curve is multiplied by the factor, K_c, where:

$$K_c = 1.0 \text{ for } S_n \leq 3S_m$$

$$K_c = 1.0 + \frac{1-n}{n(m-1)}\left(\frac{S_n}{3S_m} - 1\right) \text{ for } 3S_m < S_n \leq 3mS_m$$

$$K_c = \frac{1}{n} \text{ for } S_n \geq 3mS_m$$

where: n = 0.3 and m = 1.7 for stainless steel.

Since the stress intensity (S_n) was determined as 49,848 psi, the value of K_c to be used in the fatigue analysis is:

$$K_c = 1.0 + 3.333\left(\frac{49{,}848}{34{,}440} - 1\right) = 2.491$$

(c - f) The cumulative fatigue usage will be determined here using the K_c value where necessary, and must have a value below 1.0. The nozzle meets thermal ratcheting requirements. The maximum temperature will remain below 800°F.

Also, 304 stainless steel has adequate yield strength to ultimate strength ratio.

Cumulative Usage Factor

$$U = U_1 + U_2 + U_3 + U_4 + U_5 + U_6 + U_7$$

$$U = 0.385 + 0.032 + 0.0006 + 0.0004 + 0.013 + 0.278 + 0.002 = 0.711$$

EMCB-07-0072

In addition to Table 2.2.3-3, various components listed in Tables 2.2.2.3-1, 2.2.2.5.2.2-1 and 2.2.2.7.2-2 of LAR Attachment 5, which contain stress summaries, have failed to meet the NB-3222.2 primary plus secondary stress intensity requirement of $3S_m$. Attachment 5 states that these components have been qualified by passing the simplified elastic-plastic analysis of NB 3228.5.

a) Provide a summary of the evaluations which shows that the special rules for exceeding $3S_m$ as provided by (a) through (f) of subparagraph 3228.5 have been met.
b) Tables 2.2.2.5.2.2-1 and 2.2.2.7.2-2 also provide acceptability of components, that failed to meet the $3S_m$ allowable, through NB-3228.3. Discuss the basis and show that you meet the requirements for using the NB-3228.3 criteria. Also provide a summary of the analysis results, which shows that the requirements of NB-3228.3 have been met.

DNC Response

Several steam generator and pressurizer locations have maximum stress ranges that exceed the $3S_m$ limit in NB-3222.2. Most of these sections meet the simplified elastic-plastic analysis criteria in NB-3228.3 of the ASME B&PV Code, Section III, 1971 Edition through the summer 1973 Addendum (equivalent to NB-3228.5 in latter Code Editions). In NB-3228.3 there are six requirements, (a) through (f), which are satisfied. A summary showing that each of these requirements have been satisfied will be provided. In addition, those sections that exceed $3S_m$ and that were qualified by full elastic-plastic analysis will also be summarized in the response showing details of the plasticity analysis. Documentation of the final results of the elastic-plastic analysis is under development. A summary of the results will be provided by February 28, 2008.

EMCB-07-0073

Section 1.2 identifies that the current thermal design flow was maintained for the analyses of the six SPU cases summarized in Tables 1-1 and 1-2. Section 2.2.3.2.4 indicates that the design core bypass flow is maintained for the SPU conditions. Section 2.2.3.2.3 contains a paragraph titled "Flow-Induced Vibrations" in which it states that "The results of FIV analyses for the MPS3 SPU are provided in Table 2.2.3-1 and Table 2.2.3-2." Provide an explanation of incore changes due to SPU that would affect FIV on vessel internals and core support structures.

DNC Response

The incore changes due to SPU that would affect FIV or vessel internals and core support structures are core inlet and outlet temperature change. The FIV assessment was performed according to analytical and experimental formulations relating core inlet and outlet temperature change, which may affect the amplitude of the response and consequently the maximum stress (strain) range. The reactor core inlet and outlet temperature changed because of SPU at MPS3.

EMCB-07-0074

For FIV on the reactor internals, Section 2.2.3.2.3 states that "Based on the analysis performed for MPS3, reactor internals response due to FIV is extremely small and well within the allowable based on the high cycle endurance limit for the material. The results of FIV analyses for the MPS3 SPU are provided in Table 2.2.3-1 and Table 2.2.3-2."

a) Describe the methodology and acceptance criteria for assessing FIV on vessel internals.
b) In Tables 2.2.3-1 and 2.2.3-2, include other reactor internals susceptible to FIV such as, lower internals assembly (core barrel, thermal shield support flexures, thermal shield support bolts and the dowel pins), lower support plate, upper internals guide tubes, and upper support plate. For both tables (2.2.3-1 and 2.2.3-2) provide current, SPU and allowable values.
c) Provide the basis that established the 101.5 in/in x 10^{-6} endurance limit strain in Table 2.2.3-2.

DNC Response

a) The methodology employed utilizes prior analytical and measured vibration data to recalculate new stresses (strains) after the MPS3 stretch power uprating. Scaling the structural response to FIV was performed according to analytical and experimental formulations relating such parameters as flow rate (Mechanical Design Flow) and/or temperature change, which may affect the amplitude of the response and consequently the maximum stress (strain) range.

   The ASME Code combined with measured data forms the basis for the acceptance criteria for mechanically induced stresses/strains produced by FIV.

b) The additional components that also experience vibratory loads include: 1) lower radial keys, 2) upper core plate alignment pins, 3) lower support plate and lower support columns. The vibratory response of these components, as shown below, is extremely small.

Table 2.2.3-1

Uprated Lower Internal Critical Component Stresses Due to FIV

Component	Current Maximum Alternating Stress psi	SPU Maximum Alternating Stress psi	ASME Code Endurance Limit[(a)] (high-cycle fatigue) psi
Core Barrel Flange	1,720	2,893	23,700
Core Barrel Girth Weld	4,840	8,141	23,700
Lower Radial Keys	2,060[(b)]	Small[(b)]	23,700
Upper Core Plate Alignment Pins	2,900[(b)]	Small[(b)]	23,700
Lower Support Plate	< 225 Negligible	Negligible	23,700
Lower Support Columns	< 65 Negligible	Negligible	23,700

Note:

a) Basis is ASME Code Section NB-3222[(1)] and Figure I-9.2.2[(2)], Curve A and Table I-9.2.2[(2)].

b) The current vibration load stresses are those from the 4 loop generic reactor internals component analysis. Since the stresses are extremely small as compared to the allowable of 23,700 psi, sufficient margin exist to cover any impact due to the SPU.

Table 2.2.3-2

Uprated Upper Internal Critical Component Strains Due to FIV

Component	Current Mean Strain in/in x 10^{-6}	Uprated Mean Strain in/in x 10^{-6}	Endurance Limit Strain[(3)] in/in x 10^{-6}
Guide Tubes	17.29	18.21	101.5

c) The measured strains during the hot functional test are for the 150-inch 17x17 guide tube, which can conservatively be used for the MPS3 96-inch 17x17 guide tube. The reason for

(1) ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1998 Edition with 2000 Addenda.
(2) ASME Boiler and Pressure Vessel Code, Section II, Part D, 1998 Edition with 2000 Addenda.
(3) WCAP-8766, "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan I Power Plant," May 1976.

this conservatism is that the longer guide tube would deflect more than the shorter guide tube resulting in larger strains and stresses.

It is important to note that the core (i.e., fuel assemblies) is not present during the hot functional testing. Guide tube test results show that the strain measurements of the guide tubes without the core during hot functional testing are more than the measurements with the core present.

EMCB-07-0075

Section 2.2.2.5.2.5 indicates that the effects of SPU on the fluid-elastic stability ratio and amplitudes of tube vibration due to turbulences including vortex shedding have been accounted for. Section 2.2.2.5.2.5 also states that the analysis of the MPS3 Model F SGs indicates that significant levels of tube vibration do not occur from either the fluid-elastic, vortex shedding or turbulent mechanisms as a result of the SPU conditions. It also states that the turbulence would increase by as much as 49.6 percent, which will result in induced amplitude of 2 mils. Show quantitatively that the additional induced tube bending stresses have been accounted for and are acceptable. Provide an evaluation of FIV including fluid-elastic stability, turbulent and vorticity effects on tubes.

DNC Response

Tube bending stresses reported in the original flow-induced vibration (FIV) analysis are less than 200 psi. The additional stress induced by the SPU, using the maximum anticipated response increase, would be less than 300 psi. This increase in stress is not significant and will not result in any additional fatigue usage on the tubes since the FIV stress effects are approximately two orders of magnitude below the endurance limit for the material.

The fluid-elastic stability ratio reported in the original analysis is 0.5. Results of the SPU evaluation would conservatively predict a 23 percent increase in this response. This would result in a stability ratio of 0.62, which is less than the allowable of 1.0.

Turbulence induced displacements, originally 2 mils, are expected to be less than 3 mils in the uprated condition. This maximum displacement is less than the 146 mils needed to close the gap between tubes and cause wear.

The result of vortex shedding has been shown through experimental studies reported in the open literature to have no significant periodic wake shedding resonance for tubes inside a tube bundle in either liquid or two-phase flow conditions. Westinghouse flow tests of the tube bundle inlet region have demonstrated that tube vibration due to vortex shedding was not detectable. There are several reasons why vortex shedding would not cause detectable vibrations in the Westinghouse steam generator tube bundle:

1. Flow turbulence in the downcomer and tube bundle inlet region inhibit the formation of Von Karman's vortex trains.

2. The spatial variation of cross flow velocities along the tube precludes vortex shedding at a single frequency.

3. Both axial and cross flow velocity components exist on the tubes. The axial flow component disrupts the Von Karman vortices.

Therefore, vortex shedding is not an issue with these steam generators and need not be addressed.

EMCB-07-0076

Section 2.2.2.5 provides an evaluation summary at critical locations of the primary and secondary side SG components in Tables 2.2.2.5.2.2-1 and 2.2.2.5.2.2-2. The results indicate that, at several critical locations, the fatigue cumulative usage factor limit is determined to be very close to the allowable value of 1.0.

a) Confirm that the fatigue evaluations have been carried out to the 60 year plant life in accordance with the plant licensing renewal operating license.
b) Provide a description of the analytical evaluation (including cycles considered) for components with a fatigue usage factor greater than 0.90.
c) In addition, discuss the fatigue monitoring and/or other mitigating measures relative to the secondary manway bolts and any other locations where the calculated fatigue limit does not meet the 60-year design plant life limit.

DNC Response

a) The plant license renewal applies the existing number of occurrences for the 40-year design transients to operation for 60 years. Therefore, while now applicable to 60 years of operation, the total number of cycles used in the original fatigue analysis remains unchanged and the original fatigue analysis remains applicable.

b) The original analysis performed for the MPS3 Model F steam generators considers all of the thermal transients and cycles specified in the plant's design basis. The transients are grouped based on transient profile and the total number of cycles for all of the transients within the group is applied to the thermal analysis results for the worst transient within the group, thereby providing a conservative stress for all of the transient within the group. The fatigue analysis uses the results of the thermal analysis for all of the groups to calculate the total fatigue usage factor for that component for various stress paths through the component.

   The SG evaluation for the SPU conditions uses a factor based on a ratio of the pressure differential change, primary-side components, or pressure/temperature difference, secondary-side components. This factor is applied to the stress range for primary-side components or as an increase in pressure stress to secondary-side components. Some secondary-side components are affected as the result of a change in fluid temperature such as the feedwater system. In that case, a factor is also applied to the stress range.

It should be noted that if calculations indicate a decrease in the stress range due to the uprate, no credit was taken and the range remained unchanged from the original analysis value. Where there is a factor calculated due to an increase in the pressure differential, this factor is applied to the entire stress range, thermal and pressure stress, and not just to the pressure component of the stress. This provides for a conservative evaluation of the effects of the SPU. Since all of the transients and transient combinations used in the original analysis are used in the SPU evaluation, all of the design transients and transient cycles are considered in the SPU evaluation as defined in the Design Specification 953455, Revision 6.

c) The analysis assumes that the transients are evenly distributed over the 60-year operation of the steam generator. This is reasonable since the most significant transient combination for fatigue usage is heatup and cooldown. The replacement recommendation is based on a fatigue usage that is under 1.0 thereby, allowing margin.

The secondary manway bolts are the only components not meeting the 60-year design plant limit. This issue has been identified in the Station Corrective Action Program to ensure the bolts are replaced prior to reaching the end of their fatigue life.

<u>EMCB-07-0077</u>

Section 2.2.2.5.2.2, Structural Integrity Evaluation, states that: "The SG internal components, other than the U-tubes, are not part of the pressure boundary and, therefore, are not governed by the ASME Code. However, ASME Code Section III, Subsections NB and NF were adopted as guidelines for performing the structural analysis of these components. These components were reviewed and it was determined that they satisfy the ASME Code requirements for components not requiring an analysis for cyclic operation. As a result, a fatigue analysis was not performed for the internals. The feedwater ring was analyzed for fatigue since it is the most highly loaded of all the internals due to rapid feedwater flow and temperature changes."

a) Provide a summary of the evaluation which shows that a fatigue evaluation for the internals is not required.
b) Provide a summary of the analytical evaluation for the internals (including flow distribution baffle, steam dryer and flow-reflector) and their supports.
c) Provide a summary of the analytical evaluation for the feedwater ring that includes stresses, CUFs and allowable values.

d) Identify the Code and Code edition for the evaluation of the proposed SPU. If different from the Code of record, provide justification.

e) Provide an evaluation of FIV of the steam dryer, dryer supports and flow-reflector with respect to the fluid-elastic instability, acoustic loads and vortex shedding due to steam flow for the SPU.

DNC Response

a) The secondary-side components that are identified as non-pressure boundary are identified in the MPS3 steam generator design specification 953455, Revision 6 as non-nuclear safety (NNS). These components, while required to meet industry codes and standards, are not required to meet ASME Class 1 standards and, therefore, do not require a fatigue evaluation. For components that are attached to the primary or secondary-side pressure boundary, a Class 1 fatigue analysis is performed. While not required for the Class 2 portion of the pressure boundary, it provides additional assurance of the integrity of the secondary-side pressure boundary.

b) The evaluation of the internals gave analytical attention to the response of the internals and their interaction with the tubes for the operating conditions defined in the design specification. Not all of the internals are analyzed; only those components which are considered critical are analyzed. Critical components are defined as those whose failure (large deformation or rupture) under significant loading might violate the primary-to-secondary pressure boundary (the tubes). The non-critical components and also the loads on these non-critical components are insignificant during operation of the steam generator. Classification of critical and non-critical internals is given below:

Critical Components:

Baffle and tube support plates
Wrapper and wrapper support system
Stayrods and spacer pipes
Blowdown pipe
Lower deck plate

Non-Critical Components:

All of the components associated with the moisture separating equipment and those located above the lower deck plate.

Items attached to the pressure boundary are analyzed for the Normal and Upset conditions because of the thermal/pressure cycling or the reversed loading associated with the Operating Basis Earthquake (OBE). All other critical internals are not analyzed for the Normal and Upset conditions, because, during these conditions, the pressure drops or any other thermal or mechanical loads acting on these components are insignificant. These components are analyzed for the OBE loading to satisfy the primary stress limits for the design condition. None of the critical components are analyzed for the emergency and testing conditions because the pressure drops during the emergency conditions are negligible. Also, during testing conditions, the secondary side of the steam generators is iso-thermal and iso-baric, resulting in negligible mechanical loads.

Except for those components attached to the SG shell, a failure of these components will not lead to rupture of the primary or secondary pressure boundaries. Therefore, the ASME Code Section III rules were used as guidelines to evaluate stress in the internals.

c) The feedwater ring is loaded by seismic accelerations, by thermal transients, and feedwater nozzle rotations as a result of pipe loads. The evaluation of the load input was performed using finite element analysis techniques to determine the critical locations along the feedwater nozzle, J-nozzle, and steam generator shell. The feedring and shell feedring support attachment point were evaluated to ASME Code Section III criteria, including a fatigue evaluation.

   Results of the analysis were reported for the two critical feedring locations, at the J-nozzle weld and at the SG shell. Results of the analysis were compared to Class 1 allowables with results for all locations showing stress ratios less than 0.9 for all components and fatigue usage less than 0.85 for the feedring and J-nozzle, and 0.01 at the steam generator shell.

d) Analysis was performed to the Code of Construction, ASME B&PV Code, Section III, 1971 Edition, Summer 73 Addendum.

e) There was no FIV analysis performed for the steam dryer, dryer supports and flow-reflector. These are relatively stiff components and experience has shown that these components are sufficiently stiff that vibration effects due to steam flow are not an issue.

EMCB-07-0078

Discuss in detail the method for avoiding adverse flow effects during power ascension and after achieving SPU conditions. Include systems to be monitored, data to be collected and methods of data collection. Specify hold points and duration, inspections, plant walkdowns, vibration data locations, and planned data evaluation.

DNC Response

The MPS3 Power Ascension and Testing Plan is described in Section 2.12 of the Licensing Report. This testing plan will demonstrate that changes made to MPS3 hardware and instrumentation and control systems have been properly designed and implemented, and that MPS3 can be safely operated at the SPU power level. Implicit in the SPU power ascension test plan is the demonstration that the engineering calculations are correct and the completed analyses bound SPU operation. The MPS3 SPU test plan will confirm satisfactory performance for low power physics testing and full power operation at SPU power level, and demonstrate that all design criteria are satisfied. Following the completion of post refueling low power physics testing, power ascension testing will be conducted to ensure MPS3 can be safely operated at the SPU power level.

SPU implementation will result in higher flow rates for piping systems within the main power cycle. Secondary system piping and supports have been evaluated for the higher steam and feed flow rates. The evaluations concluded that the piping systems remain acceptable and will continue to satisfy design basis requirements. Vibration monitoring and system walk downs will be performed during the power ascension to the SPU level. This will ensure secondary piping system and component vibrations remain acceptable at the higher power level.

There are no MPS3 RCS (primary side) mass or volumetric flow rate changes. Flow induced vibration at SPU conditions was evaluated for the reactor vessel internals and steam generator tubes. The proposed SPU does not adversely impact the reactor vessel internals structural integrity. Operation at the higher power level will not result in rapid rates of steam generator tube wear or high levels of tube vibration to the general tube population. Therefore, vibration issues on the plant primary side are not expected.

The Power Ascension Test Procedure, which is currently under development, will be used during the return of MPS3 to power operation after the Fall 2008 refueling outage. The Power Ascension Test Procedure will be integrated with existing plant procedures to provide additional administrative controls as MPS3 power level is increased to the new rated thermal power level of 3650 MWt. It will provide operational guidance for the power increase, and direct the monitoring of plant systems, components, and parameters to ensure safe plant operation. It will also contain verification steps to ensure all of the required plant modifications have been completed and properly retested. The Power Ascension Test Procedure is expected to contain the following aspects.

Prior to plant shutdown for 3R12, baseline data will be collected for the systems and components that will be affected by the proposed power increase to 3650 MWt. The data will be collected at the current full power level (3411 MWt). If plant conditions permit, data will also be collected at a lower power level during the plant shut down for refueling. The data collected will be used to develop the expected values for the plant parameters that will be monitored as power is increased to the new full power level (3650 MWt) after the refueling outage. Data parameters will be classified as a Level 1 or Level 2 parameter. A Level 1 parameter will be associated with a Technical Specification or a critical plant parameter. A Level 2 parameter will be associated with expected system transient performance whose characteristics can be improved by equipment adjustments. Specific guidance will be provided for the required actions if either a Level 1 or Level 2 parameter deviates from an expected value during the power ascension.

The baseline data to be collected is expected to include system parameters (e.g., pressure, temperature, flow, and level), secondary component and piping vibrations, controller settings, valve positions, motor amps, and radiation levels for the systems, components and areas listed below.

Condensate System
- Main Condenser
- Condensate Pumps
- Condensate Demineralizers

Feedwater System
- Main Feedwater Pumps
- Main Feedwater Level Control Valves

Heater Drain System
- Low Pressure Feedwater Heaters
- Low Pressure Feedwater Heater Level Control Valves
- Heater Drain Pumps
- High Pressure Feedwater Heaters
- High Pressure Feedwater Heater Level Control Valves

Extraction Steam System
Moisture Separator Reheater and Drain System
- Moisture Separator Drain Pumps
- Moisture Separator Drain Tanks
- Moisture Separator Drain Tank Level Control Valves
- Moisture Separator Reheat Tank Level Control Valves

Main Turbine
- Turbine Lube Oil
- Electrohydraulic Control Oil
- Main Turbine Parameters

Main Generator
- Hydrogen Cooling
- Bus Duct Cooling
- Main Generator Parameters

Main Transformers
- Cooling System Operation
- Main Transformer Parameters

Main Steam System
Steam Generator Blowdown System
Circulating Water System
Turbine Plant Component Cooling Water System
Reactor Coolant System

- Pressurizer
- Pressurizer Level Control
- Reactor Coolant Pumps
- Nuclear Core Parameters

Reactor Plant Chilled Water System
Containment

- Containment Air Recirculation Cooling Fans
- Containment Penetration Coolers

Area Temperatures

- Containment
- Main Steam Valve Building
- Turbine Building

Radiation Surveys

- Auxiliary Building
- Engineered Safety Features Building
- Main Steam Valve Building

During the power escalation after the refueling outage, numerous hold points will be identified (e.g., 30%, 50%, 75%, 93%, 97%, 99% and 100%). Power will be gradually increased to each of the specified hold points. After power has been stabilized, plant personnel will perform system and area walk downs while collecting the required data using the plant computer, installed instrumentation, and test equipment. Proper system operation will be verified by comparing the data collected to the expected values. Any deviations outside of the expected range will be evaluated before the power increase is allowed to continue. The periodic collection of data, and the verification that the data is within the expected ranges will ensure no adverse plant operating conditions (e.g., adverse flow effects) develop during the power ascension to 3650 MWt. An additional set of data will be collected after the new full power is reached. The evaluation of this final set of data will verify that MPS3 can safely operate at a rated thermal power of 3650 MWt.

EMCB-07-0079

Discuss the evaluation of potential FIV effects due to the increase in steam flow resulting from SPU conditions. The evaluation should include the SG internals, steam and feedwater systems and their associated components. Include impact on structural capability and performance during normal operations, anticipated transients (initiation and response), and design-basis conditions. Discuss procedure in place for preparation and response to the potential occurrence of loose parts as a result of the SPU. The evaluations should also include calculations, when applicable, of the fluid-elastic stability ratio, and stresses due to turbulent and vortex shedding.

DNC Response

Steam and feedwater systems and their associated components will be assessed for potential effects of FIV in accordance with the overall plan for piping vibration and in accordance with ASME OM Part 3. (See EMCB-07-0075 and EMCB-07-0077 for steam generator internal aspects)

EMCB-07-0080

Provide a summary of the evaluation of thermowells in the main steam (MS), feedwater (FW) and Condensate piping systems for increased vibrations due to the increased SPU flow rate.

DNC Response

The thermowells installed in the condensate, feedwater and main steam systems are designed for the maximum velocities listed below:

Water systems	30 ft/sec.
Steam systems	300 ft/sec.

As part of the SPU evaluations, the velocities in each system were calculated. The maximum velocity in the main steam piping was calculated to be 264 feet per second, which is below the 300 feet per second maximum design velocity for thermowells in steam systems. The maximum velocity in a line which contains a thermowell in the condensate and feedwater systems is 18.24 feet per second in the feedwater pump discharge piping, which is below the 30 feet per second design velocity for water systems. The SPU velocities are lower than the design velocities for thermowells and; therefore, they are acceptable for the increased flow and potential increased vibration.

The condensate, feedwater and main steam systems do not contain any sample probes extending into the flow stream. Sampling is performed by the turbine plant sampling system which has the capability to continuously monitor for ammonia, pH, chloride, oxygen, sodium, and conductivity and the capability to take corrosion product samples. The samples are obtained via socket welded connections to the piping systems, reduced in pressure, and cooled as required for analysis.

EMCB-07-0081

Section 2.2.2.4, Control Rod Drive Mechanism (CRDM), identifies that the CRDM was designed in accordance with the ASME B&PV Code, Section III, Division 1, 1974 edition through summer 1974 addenda, for normal, upset, emergency and faulted conditions. It also contains a summary of the results of the evaluations performed for the SPU which is presented in Table 2.2.2.4-1 through 2.2.2.4-5.

a) Confirm that the fatigue evaluations have been carried out to the 60-year plant life in accordance with the plant licensing renewal operating license.
b) In Table 2.2.2.4-1 through 2.2.2.4-4, provide corresponding values for the current licensed power.
c) In Table 2.2.2.4-1 (upper joint), for the Bell mounting threaded area, provide the basis of the allowable values of 19,479 psi and 21,755 psi for the normal and upset conditions respectively. Also, provide the component material designation in the first column of Table 2.2.2.4-1 through 2.2.2.4-3. The allowable values (shown for design temperature) for the Bell mounting threaded area are different for the upper joint, the middle joint and the lower joint. Provide a justification for the difference in allowable values.
d) Table 2.2.2.4-3 (lower joint) in the first column in the upper hand corner is marked "Middle Joint". Verify that the stress summary of Table 2.2.2.4-3 is for the Lower Joint components.

DNC Response

a) The fatigue evaluations have been performed to the 60-year plant life in accordance with the plant licensing renewal operating license.

b) Table 2.2.2.4-1 through Table 2.2.2.4-4 were revised to include the values for the current licensed power (Design Condition). The tables are provided following the response to this question. Since most of the values listed in Tables 2.2.2.4-1 through 2.2.2.4-4 are unchanged from the current licensed power, only the changed values will be noted and accompanied by the current value.

c) The allowable values in the upper joint bell mouthing area are based off the local temperature where the max stresses occurred. Per the analysis of record (AOR), since the stresses are high in this region, the local temperatures were used to eliminate unnecessary conservatism. The local temperatures 476°F and 329°F (found in the AOR, EM-4531, Rev. 2) correspond to the yield stresses 19,720 and 21,970 (these were further reduced to 19,479, and 21,755 due to the uprating). The reason these allowable values differ from those used for the middle and lower joint is because those two are based on the design temperature 650°F.

   The component material designation was added in the first column of tables 2.2.2.4-1 through 2.2.2.4-3.

d) The stress values on Table 2.2.2.4-3 are the correct values for the lower joint. Please replace "Middle Joint" with "Lower Joint" in the first column of the table header. The header for Table 2.2.2.4-3 was mistakenly carried over from Table 2.2.2.4-2.

Table 2.2.2.4-1
Upper Joint Components' Stress Summary

Upper Joint		Design Condition		Normal Condition		Upset Condition		Testing Condition		***Special Condition***		Faulted Condition	
Component	**Param. Per ASME Code III**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**
Cap (SA479 304)	P_m	5,954 Note 3	16,100 Note 3					7,400 Note 2 (5,994)	16,110 Note 3			7,216 Note 2 (5,994)	19,320 Note 3
	P_m+P_b	20,757 Note 3	24,150 Note 3					22,212 Note 2 (20,757)	24,165 Note 3			22,028 Note 2 (20,757)	28,980 Note 3
	P_m+P_b+Q			19,107 Note 3	48,300 Note 3	19,128 Note 3	48,300 Note 3						
	$\sigma_1+\sigma_2+\sigma_3$									-16,522 Note 3	64,400 Note 3		
Rod Travel Housing (SA336 F8)	P_m	14,172 Note 3	16,100 Note 3					17,613 Note 2 (14,172)	21,420 Note 2 (16,110)			17,176 Note 2 (14,172)	19,320 Note 3
	P_m+P_b	19,419 Note 3	24,150 Note 3					20,826 Note 2 (17,385)	32,130 Note 2 (24,165)			20,389 Note 2 (17,385)	28,980 Note 3
	P_m+P_b+Q			23,574 Note 3	48,300 Note 3	21,106 Note 3	48,300 Note 3						
	$\sigma_1+\sigma_2+\sigma_3$									13,922 Note 3	64,400 Note 3		

Canopy (SA336 F8 and SA479 304)	P_m	4,606 Note 3	16,100 Note 3					5,724 Note 2 (4,606)	16,110 Note 3			5,582 Note 2 (4,606)	19,380 Note 3
	P_m+P_b	8,254 Note 3	24,150 Note 3					9,372 Note 2 (8,254)	24,165 Note 3			9,230 Note 2 (8,254)	28,980 Note 3
	P_m+P_b+Q			27,594 Note 3	48,300 Note 3	40,057 Note 3	48,300 Note 3						
	$\sigma_1+\sigma_2+\sigma_3$									9,667 Note 3	64,400 Note 3		
Threaded Area (SA336 F8 and SA479 304)	Pm (Shear)									5,370 Note 3	9,660 Note 3		
	2x Shear									38,020 Note 3	48,300 Note 3		
	P_m+P_b+Q									47,500 Note 3	48,300 Note 3		
	Bell Mouthing Stress Intensity			19,639 Note 1, 2 (21,022)	19,479 Note 1, 2 (19,720)	20,187 Note 2 (21,106)	21,755 Note 2 (21,907)						

Note 1: This stress exceeds the allowable by 160 psi. This is considered acceptable due to the conservatism that the maximum design temperature of 650°F was used, as opposed to the hot leg temperature of 622.6°F, for the hot boundary of the steady state transient. The ASME Code allowable yield strength, Sy, is 19,479 psi at the nodal temperature of 494°F. Reducing the nodal temperature by the ratio (622.6/650) to 473°F yields an allowable Sy of 19,749 psi.

Note 2: New stress per uprate analysis. Stress corresponding to current licensing power is listed in parentheses ().

Note 3: Stress is not impacted by uprate.

Note 4: Shaded sections indicate inapplicability.

Table 2.2.2.4-2
Middle Joint Components' Stress Summary

Middle Joint		**Design Condition**		**Normal Condition**		**Upset Condition**		**Testing Condition**		**Special Condition**		**Faulted Condition**	
Component	**Param. Per ASME Code III**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**
Rod Travel Housing (SA336 F8)	P_m	6,288 Note 2	16,100 Note 2					7,815 Note 1 (6,288)	16,110 Note 2			7,621 Note 1 (6,288)	19,320 Note 2
	P_m+P_b	8,172 Note 2	24,150 Note 2					9,669 Note 1 (8,172)	24,165 Note 2			9,505 Note 1 (8,172)	28,980 Note 2
	P_m+P_b+Q			16,669 Note 2	48,300 Note 2	14,388 Note 2	48,300 Note 2						
	$\sigma_1+\sigma_2+\sigma_3$									-14,654 Note 2	64,400 Note 2		
Latch Housing (SA351 CF8)	P_m	11,930 Note 2	15,300 Note 2					14,827 Note 1 (11,930)	15,300 Note 2			14,459 Note 1 (11,930)	18,360 Note 2
	P_m+P_b	15,659 Note 2	22,950 Note 2					18556 Note 1 (15,659)	22,950 Note 2			18,188 Note 1 (15,659)	27,540 Note 2
	P_m+P_b+Q			17,431 Note 2	45,900 Note 2	16,395 Note 2	45,900 Note 2						
	$\sigma_1+\sigma_2+\sigma_3$									15056 Note 2	61,200 Note 2		

Canopy (SA336 F8 and SA351 CF8)	P_m	4,460 Note 2	15,300 Note 2					5,543 Note 1 (4,460)	15,300 Note 2			5,406 Note 1 (4,460)	18,360 Note 2
	P_m+P_b	6,844 Note 2	22,950 Note 2					7,927 Note 1 (6,844)	22,950 Note 2			7,790 Note 1 (6,844)	27,540 Note 2
	P_m+P_b+Q			45,504 Note 2	45,900 Note 2	38,164 Note 2	45,900 Note 2						
	$\sigma_1+\sigma_2+\sigma_3$									5439 Note 2	61,200 Note 2		
Threaded Area (SA336 F8 and SA351 CF8)	Pm (Shear)									3314 Note 2	9,180 Note 2		
	2x Shear									11272 Note 2	45,900 Note 2		
	P_m+P_b+Q									31100 Note 2	45,900 Note 2		
	Bell Mouthing Stress Intensity			14,136 Note 2	17,000 Note 2	11,069 Note 2	17,000 Note 2						

Note 1: New stress per uprate analysis. Stress corresponding to current licensing power is listed in parentheses ().

Note 2: Stress is not impacted by uprate.

Note 3: Shaded sections indicate inapplicability.

Table 2.2.2.4-3
Lower Joint Components' Stress Summary

Lower Joint		**Design Condition**		**Normal Condition**		**Upset Condition**		**Testing Condition**		**Special Condition**		**Faulted Condition**	
Component	**Param. Per ASME Code III**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**
Latch Housing (SA351 CF8)	P_m	12,380 Note 3	15,300 Note 3					15,386 Note 2 (12,380)	21,375 Note 2 (15,300)			15,005 Note 2 (12,380)	18,360 Note 3
	P_m+P_b	16,650 Note 3	22,950 Note 3					19,656 Note 2 (16,650)	32,062 Note 2 (22,950)			19,275 Note 2 (16,650)	27,540 Note 3
	P_m+P_b+Q			16,921 Note 3	45,900 Note 3	15,228 Note 3	45,900 Note 3						
	$\sigma_1+\sigma_2+\sigma_3$									15,560 Note 3	61,200 Note 3		

Head Adaptor (SA182 304)	P_m	7,343 Note 3	16,100 Note 3					9,126 Note 2 (7,343)	16,100 Note 3			8,900 Note 2 (7,343)	19,320 Note 3
	P_m+P_b	10,070 Note 3	24,150 Note 3					11,853 Note 2 (10,070)	24,165 Note 3			11,627 Note 2 (10,070)	28,980 Note 3
	P_m+P_b+Q			15,165 Note 3	48,300 Note 3	13,467 Note 3	48,300 Note 3						
	$\sigma_1+\sigma_2+\sigma_3$									15,824 Note 3	64,400 Note 3		
Canopy (SA182 304 and SA351 CF8)	P_m	9,345 Note 3	15,300 Note 3					11,614 Note 2 (9,345)	15,300 Note 3			11,326 Note 2 (9,345)	18,360 Note 3
	P_m+P_b	19,011 Note 3	22,950 Note 3					21,280 Note 2 (19,011)	22,950 Note 3			20,992 Note 2 (19,011)	27,540 Note 3
	P_m+P_b+Q			45,985 Note 1, 3	45,900 Note 3	37,560 Note 3	45,900 Note 3						
	$\sigma_1+\sigma_2+\sigma_3$									28,702 Note 3	61,200 Note 3		

Threaded Area (SA182 304 and SA351 CF8)	Pm (Shear)									4,103 Note 3	9,180 Note 3		
	2x Shear									12,852 Note 3	45,900 Note 3		
	P_m+P_b+Q									33,200 Note 3	45,900 Note 3		
	Bell Mouthing Stress Intensity			13,733 Note 3	17,000 Note 3	9,720 Note 3	17,000 Note 3						

Note 1: This stress exceeds the allowable by 85 psi. This is considered insignificant due to the conservatism that the allowable is based on the design temperature of 650°F as opposed to the actual nodal temperature of 76°F. The ASME Code allowable stress intensity S_m is 20 ksi at 78°F and 15.3 at 650°F.

Note 2: New stress per uprate analysis. Stress corresponding to current licensing power is listed in parentheses ().

Note 3: Stress is not impacted by uprate.

Note 4: Shaded sections indicate inapplicability.

Table 2.2.2.4-4 Cumulative Fatigue Usage Factors for CRDM Joints

Joint	**Component**	**Total Usage Factor**		**Allowable Usage Factor**
		Current Analysis	**SPU Analysis**	
UPPER	Cap	**0.0**	0.0	1.0
	Road Travel Housing	**0.0**	0.0	1.0
	Canopy	**0.938**	0.491	1.0
	Weld Canopy	0.511	**0.527**	1.0
	Threaded Area	**0.362**	0.234	1.0
MIDDLE	Road Travel Housing	**0.0**	0.0	1.0
	Latch Housing	**0.0**	0.0	1.0
	Canopy	**0.0**	0.0	1.0
	Weld Canopy	**0.524**	0.013	1.0
	Threaded Area	0.000	**0.039**	1.0
LOWER	Latch Housing	**0.0**	0.0	1.0
	Head Adaptor	**0.0**	0.0	1.0
	Canopy	0.000	**0.011**	1.0
	Weld Canopy	0.0243	**0.027**	1.0
	Threaded Area	0.000	**0.031**	1.0

Note: Values in bold represent the bounding usage factors. All bounding values are less than the allowable usage factor of 1.0; therefore they are acceptable.