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January 11, 2008

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

Serial No.: 07-0834E
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 SBPB-07-0082 THROUGH SBPB-07-0087

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 27, 2007 letters. DNC responded to the RAIs in letters dated November 19, 2007 (Serial No. 07-0751) and December 17, 2007 (Serial No. 07-0499). The NRC staff forwarded an additional RAI in a December 14, 2007 letter. The response to questions SBPB-07-0082 through SBPB-07-0087 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 11TH day of January, 2008.

My Commission Expires: May 31, 2010.

Vicki L. Hull
Notary Public

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2010

Commitments made in this letter: None

Attachment

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT

LICENSE AMENDMENT REQUEST

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RESPONSE TO QUESTIONS SBPB-07-0082 THROUGH SBPB-07-0087

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

Balance-of-Plant Branch

SBPB-07-0082

In Attachment 5, Section 2.5.6.3, Solid Waste Management Systems, the licensee states "Implementation of SPU is anticipated to increase the potential for occurrence of the crud induced power shift (CIPS) phenomena. Details associated with the fuel cleaning process proposed to manage and/or preclude CIPS require finalization." Consistent with the requirements of 10 CFR 50.34a(c), describe any new equipment necessary for control of liquid effluents from the cleaning process and the effect that treatment of those effluents would have on the packaging and storage of solid waste.

DNC Response

Dominion Nuclear Connecticut (DNC) is currently in the process of evaluation and selection of the vendor that will provide the fuel cleaning services. As such, no details about the equipment are available at this time.

The fuel cleaning systems currently under evaluation do not include any permanently installed equipment and would not involve the creation of any liquid radioactive effluents. The crud removed from the fuel will be collected on filters and the filters would require subsequent disposal. It is expected that these filters would not be significantly different from other filters that collect radioactive materials.

The normal change processes will be applied when the fuel cleaning equipment is placed into service (e.g. procedure changes and temporary modifications). These change processes will insure that the control of liquid and solid radioactive wastes will comply with all applicable regulations and requirements.

SBPB-07-0083

In Attachment 1, Section 3.3, Demineralized Water Storage Tank (DWST) Licensing Basis Change, the licensee proposes to change the licensing basis for the required level in the DWST. The existing basis is for the DWST to hold enough water for 10 hours at hot standby. The licensee proposed to change the basis to hold enough water for 7 hours at hot standby. To satisfy accident analysis assumptions, the DWST must contain sufficient cooling water to remove decay heat following a reactor trip, and then to cool down the reactor coolant system (RCS) to residual heat removal entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. Provide an evaluation of the proposed licensing basis change, including the basis for the current 10 hour requirement and the basis to conclude that the proposed seven hours adequately addresses all accident analyses and requirements.

DNC Response

1. *General*

Regarding cold shutdown capabilities, the MPS3 DWST design and licensing basis considered, but does not comply with, Branch Technical Position (BTP) RSB 5-1 (see FSAR Section 10.4.9, "Auxiliary Feedwater System"). Furthermore, BTP RSB 5-1 doesn't require the functional capability for cold shutdown (with crediting only safety-grade SSC's) for all accidents. This approach would have been the DWST design and licensing basis if MPS3 fully complied with draft Regulatory Guide (RG) 1.139, "Guidance for Residual Heat Removal" (May 1978) which contained the following sentence:

> "Consequently, it is essential that a power plant have the capability to go from hot-standby to cold-shutdown conditions (when this is determined to be the safety course of action) under any accident conditions."

R.G. 1.139 was never issued by NRC; but rather, BTP RSB 5-1 was issued and it defined a method acceptable to the NRC for complying with GDC 19 and 34. BTP RSB 5-1 Position A specifies a functional capability for the reactor to be taken "from normal operating conditions" to cold shutdown using only safety grade systems.

As indicated in FSAR 5.4.7.2.3.5, "Safety Grade Cold Shutdown", the design Safety Grade Cold Shutdown (SGCS) assumes a safe shutdown earthquake (SSE) coincident with a loss of offsite power. The postulated seismic event may impact alternate non-seismic auxiliary feedwater (AFW) pump suction sources or impact longer-term DWST replenishment activities/capabilities.

License Report Table 2.5.4.5-2, "Demineralized Water Storage Tank – SGCS Functional Requirements Comparison Before and After Uprate" provides an assessment of MPS3 SGCS functional requirements.

The DWST is required to have sufficient inventory to mitigate numerous accidents, but in general FSAR Chapter 15 accidents are mitigated relatively quickly. For example, the licensing basis for a main steam line break (MSLB) accident is hot-standby and this accident is mitigated using only a small portion of the DWST inventory. A more detailed accident assessment is provided in Section 3 of this response.

MPS3 SPU License Amendment Request, Attachment 1, page 28 provides a technical analysis for the DWST licensing bases change and it contains the following statement:

> "The proposed licensing basis change provides adequate inventory for accident analysis primary success paths and provides adequate inventory which operating experience and/or probabilistic risk assessment has shown to assure public health and safety."

2. *Bases for Current DWST 10/6 Design Criterion*

The 10-hour at hot-standby, followed by a 6-hour cooldown to RHR entry conditions (10/6) DWST design criterion for MPS3 was established early in the licensing process. The MPS3 PSAR (circa 1973) describes a 200,000-gallon DWST (which would have been equivalent to an approximate 2/5 DWST design criterion).

In 1979, the 10/6 DWST sizing criterion was selected by the A/E (in consultation with the Nuclear Steam Supply System (NSSS) vendor in expectation of a SGCS design that might be required to comply with draft Regulatory Guide (RG) 1.139 (i.e., under any accident condition). Specifically, internal correspondence shows that plant designers were contemplating a SGCS design that included all Condition II, III, IV events, including an event that required control room evacuation, which would have resulted in safe shutdown being conducted from the Auxiliary Shutdown Panel (ASP). A/E and NSSS vendor correspondence in 1979 show that the 10-hour hot-standby period was comprised of the following time intervals:

Table 1
DWST - Hot-Standby Phase Criterion Circa 1979

Design Consideration	Time (Hours)
"safety grade short-term allowance"	4
"boration period"	6
Sub-total	10

As the SGCS design evolved, including BTP RSB 5-1 issuance in lieu of RG 1.139, the outside the control room and the "under any accident conditions" design aspects were eliminated from the SGCS design. The MPS3 SGCS design and licensing bases evolved to address only a safe shutdown earthquake (SSE) and coincident Loss of Normal Power (LNP). Since the SGCS design and licensing basis doesn't involve a control room evacuation or a concurrent design basis accident that complicated operator response, the current SGCS analysis assumes operators begin boration without significant delay, if the DWST is the only available SG makeup source (i.e., a seismic event damages the Condensate Storage Tank (CST) making it unavailable as an alternate AFW pump suction source).

In summary, the DWST sizing criterion was increased to the current 10/6 criterion in 1979 based upon preliminary actions in expectation of the SGCS design being required to comply with draft RG 1.139 (i.e., the SGCS design that was evolving at that time and a design that wasn't fully implemented).

3. *Assessment of DWST Accident Mitigation Functional Requirements Against the 7/6 Design Criterion*

Table 2 provides a technical basis to conclude that the proposed DWST 7/6 design criterion adequately supports all accident mitigation requirements. Specifically, representative FSAR Chapter 15 accidents are reviewed against the proposed DWST 7/6 design criterion.

Table 2
FSAR Chapter 15 Accident Analysis Assessment Against the Proposed DWST 7/6 Design Criterion (Selected Accidents)

Accident	Assessment	7/6 design criterion
LBLOCA	Safe Shutdown Licensing Basis: N/A AFW system isn't credited for LBLOCA mitigation. Recirculation Spray (RSS) system, via cold leg and hot leg injection pathways, support the long-term cooling design function required by 10 CFR 50.46.	Acceptable
SBLOCA	Safe Shutdown Licensing Basis: N/A AFW system is credited in the SBLOCA analysis. The SBLOCA classification includes a break spectrum from 13.5 to 3/8-inch diameter. As specified in WCAP 9600, "Report on Small Break Accidents for Westinghouse NSSS System" the SBLOCA long-term cooling design function is performed by sump recirculation using emergency core cooling system (ECCS) [i.e., Recirculation Spray System (RSS) which will reject decay heat to the ultimate heat sink (UHS)]. From a design and licensing perspective, RSS system/ECCS is the credited long-term pathway and the RHR system is a	Acceptable

Accident	Assessment	7/6 design criterion
	potentially available SSC that can be used at operator discretion. The spectrum upper end (i.e., 13.5 to 2-inch diameter) is mitigated relatively quickly. The SG's don't perform any long-term heat removal function because heat removed by the break flow becomes sufficient (or within 1- hour only a very small SG steam rate is required) and the RSS System is the long-term heat removal pathway. SBLOCA's on the lower end of the spectrum require operator action to establish a long-term cooling method. If condensate makeup or other SSC availability becomes a concern, plant operators can open the steam generator atmospheric dump bypass valves (SGADBV) and/or pressurizer power-operated relief valve (PORV's) to establish the credited SBLOCA long-term cooling method (i.e., ECCS sump recirculation). If a PORV is opened to establish the credited sump recirculation long-term heat removal pathway, the SG heat removal pathway isn't required beyond this point. The smallest un-isolable SBLOCA would have an RCS leakage rate greater than 126-gpm at 2250 psia (Ref. FSAR page 9.3-56). A complex analysis would be required to determine long-term DWST usage in light of the decay heat removed by emergency core cooling system (ECCS) flow. However, DWST steaming support capability would significantly extend beyond 13-hours due to heat removed by the feed and bleed heat removal pathways. Thus, the 334,000-gallon DWST limiting condition of operation is more than adequate time to ensure the SBLOCA's credited long-term cooling method can be established, if required.	
Main Steam Line Break [FSAR 15.1.5]	Safe Shutdown Licensing Basis: Hot Standby FSAR Section 15.0 defines the Main Steam Line Break accident as a Condition IV event. Secondary side piping failure events are excluded from the SGCS design and licensing basis because subsequent cooldown is not from "normal operating conditions".	Acceptable

Accident	Assessment	7/6 design criterion
	MPS3 would be brought to cold shutdown on a schedule commensurate with safety and using available SSC's. [FSAR Section 15.1.5 shows accident is mitigated within 1-hour]	
Feedwater Line Break [FSAR 15.2.8]	Safe Shutdown Licensing Basis: Hot Standby FSAR Section 15.0 defines the Feedwater Line Break accident as a Condition IV event. Secondary side piping failure events are excluded from the SGCS design and licensing basis because subsequent cooldown is not from "normal operating conditions". MPS3 would be brought to cold shutdown on a schedule commensurate with safety and using available SSC's. [FSAR Figure 15.2.13 to 24 show accident is mitigated within 1-hour]	Acceptable
Rod Cluster Control Assembly Ejection, Spectrum of [FSAR 15.4.8]	Safe Shutdown Licensing Basis: Hot Standby This event is classified as a Condition IV event. The accident's reactivity control aspects are mitigated relatively quickly. The accident's primary system pressure boundary failure aspects would be governed by the SBLOCA analysis, which is addressed above. MPS3 would be brought to cold shutdown on a schedule commensurate with safety and using available SSC's. [FSAR Table 15.0-6 doesn't list the AFW system as a system required for accident mitigation]	Acceptable
Loss of Non-emergency AC Power to the Station Auxiliaries & Loss of Normal Feedwater with Loss of Normal Power	Safe Shutdown Licensing Basis: Cold Shutdown The Loss of Non- emergency AC Power accident is within the SGCS design basis because loss of normal power (LNP) event is encompassed in the SGCS design scenario (seismic event with concurrent loss of normal power). Therefore, the SGCS design and licensing basis encompasses this accident. The SPU's SGCS analysis demonstrates that the DWST 7/6 criterion provides adequate condensate.	Acceptable

Accident	Assessment	7/6 design criterion
[FSAR 15.2.6 and 15.2.7]	A LNP causes a loss of normal feedwater (LONF) event and FSAR Section 15.2.6 says that Section 15.2.7 addresses the loss of feedwater aspect of the loss of AC power events.	
Locked Rotor [FSAR 15.3.3]	Safe Shutdown Licensing Basis: Hot Standby This accident is mitigated relatively quickly. FSAR Table 15.0-6, "Plant System and Equipment Required for Mitigation of Transient or Accident" shows that AFW system is not credited for locked rotor mitigation. FSAR Section 15.0 defines the locked rotor accident as a Condition IV event. This accident would be outside the BTP RSB 5-1's "from normal plant conditions" criterion and this accident is outside the MPS3 SGCS design scenario (which is bases for MPS3's degree of compliance to BTP RSB 5-1). MPS3 would be brought to cold shutdown on a schedule commensurate with safety and using available SSC's.	Acceptable
SGTR [FSAR 15.6.3]	Safe Shutdown Licensing Basis: Hot Standby/Hot Shutdown AFW system is credited in the SGTR analysis. This accident involves RCS cooldown/depressurization to terminate the break flow into the impacted SG. This occurs within 2-hours and the 7/6 criterion is more than sufficient to support accident mitigation. A complex analysis would be required to determine long-term DWST usage in light of the decay heat removed by emergency core cooling system (ECCS) flow but the DWST capability would be extended by the ECCS decay heat removal pathway. FSAR Section 15.0 defines the SGTR accident as a Condition IV event. Accidents involving primary system pressure boundary passive failures are beyond the SGCS design and licensing because they don't meet the "from normal operating condition" criterion.	Acceptable

Accident	Assessment	7/6 design criterion
	MPS3 would be brought to cold shutdown on a schedule commensurate with safety and using available SSC's. [FSAR Table 15.6.3-2 shows break flow terminated within 2-hours. Offsite radiation dose for SGTR accident performed with methodology developed in WCAP 10698, Supplement 1]	

The SGCS analysis is based upon an 11-hours RHR entry time (which is discussed in the following RAI response). Not withstanding the accident safe shutdown modes, the proposed 7/6 DWST design criterion affords water for in excess of 13.7-hours for most accidents because the proposed DWST design criterion conservatively retains a feedwater line break spillage allowance (which is equivalent to 0.7-hours steaming time).

Therefore, an approximate 2.7-hour time margin is retained between the SGCS analysis RHR entry time and DWST 7/6 design criterion. Therefore, although beyond the plant design and licensing basis, it is reasonable to conclude that cold shutdown conditions can be achieved for many accidents crediting only the DWST inventory because the decision to initiate RCS boration would likely be made within 2.7-hours. As stated in the MPS3 license amendment request (Attachment 1, page 28), the DWST 7/6 design criterion also affords adequate time for DWST replenishment activities, if additional condensate is required.

4. Summary

SPU SGCS assessments have concluded that the 7/6 DWST design criterion results in ample usable water for SGCS design scenarios. The 7/6 DWST design criterion provides ample water to mitigate all FSAR Chapter 15 design basis accidents. The MPS3 licensing report addresses the station blackout and fire shutdown design requirements.

SBPB-07-0084

In Attachment 5, Section 2.8.4.4.2.2.2, Safety Grade Cold Shutdown (SGCS) Cooldown Analysis, the licensee states that TS 3.7.1.3 ensures an adequate volume in the DWST to support hot standby conditions with subsequent RCS cooldown. The licensee proposes to change the reasonable time period to 72 hours from 76 hours, but keeps the 36-hour requirement to initiation of residual heat removal. The safety related water source is the DWST, which at SPU conditions will only have the capacity for approximately 13 hours, as stated in Section 2.5.4.5, Auxiliary Feed Water. Since SGCS takes credit for only safety-related equipment, not crediting the use of the non-safety-related condensate storage tank and service water, explain how the 36-hour requirement would be maintained by safety-related equipment.

DNC Response

Branch Technical Position (BTP) RSB 5-1 (Rev. 2, July 1981) guidance does not include a 36-hour functional requirement for a safety-related steam generator (SG) inventory makeup supply. However, the regulatory position in BTP RSB 5-1, Position A provides the functional requirement to bring the reactor to a cold shutdown condition within a reasonable period of time following shutdown, assuming the most limiting single failure and using only safety-grade systems.

MPS3 is proposing the following change:

Table 1
SGCS - reasonable time period for cold shutdown

	Pre-SPU	Post-SPU
reasonable time from reactor trip to cold shutdown	66-hours	72-hours

The proposed MPS3 time was determined based on meeting the BTP RSB 5-1 referenced time from reactor trip to 200°F rather than the current reference, which is to RHR entry. MPS3 FSAR Section 5.4.7.2.5, "Safety Grade Cold Shutdown" (page 5.4-32, first full paragraph, last sentence) reads: "Therefore, the MPS3 licensing basis is to achieve cold shutdown within 66-hours of reactor trip." SRP 5.4.7, Section III.5 (review procedure section) states: "the reviewer determines that the system(s) has the capability to bring the reactor to condition permitting operation of the RHR system in a reasonable period of time, ... For the purpose of this review, 36-hours is considered a reasonable time period". Between the Standard Review Plan (SRP) and BTP, there is a frame of reference change (i.e., RHR entry time versus cold shutdown time).

Basically, the SRP 5.4.7 review procedure section is saying verify that the reactor has the capability to be brought to RHR entry conditions within 36-hours, but it never defines what is a reasonable time period between reactor trip and the cold shutdown condition. The "brought to RHR entry conditions within 36-hours" wording is used in the MPS3 SER (page 5-23).

SPU has no adverse impact upon SGCS analysis RHR entry time. Table 2 summarizes the RHR entry time reported in LR Table 2.8.4.4-5 and 2.8.4.4-6:

Table 2
SGCS Analysis – RHR Entry Time

	Pre-SPU	Post-SPU
RHR Entry Time [time after reactor trip]	11-hours	11-hours

Therefore, MPS3 continues to satisfy the within 36-hour RHR entry time criterion delineated in SRP 5.4.7, Section III.5.

Additional Information

1. MPS3 SPU License Amendment Request Section 2.8.7.2, "Natural Circulation Cooldown" reports that SPU has no impact upon SGCS boration times. Section 2.5.5.1, "Main Steam" (page 2-5-113) reports that existing steam generator atmospheric release capability continues to satisfy functional requirements inherent in the SGCS analysis. These sections support the conclusion that SPU has no impact upon SGCS analysis RHR entry time.

2. BTP RSB 5-1, Position G states:

   "The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot-shutdown for at least 4-hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure."

   The proposed 7-hour at hot-standby, followed by a 6-hour cooldown to RHR entry conditions (7/6) DWST design and licensing bases satisfies this regulatory position based upon natural circulation cooldown and the SGCS analysis. Natural circulation cooldown to RHR entry is consistent with MPS3's commitment to BTP RSP 5-1.

Summary

There is no functional requirement derived from BTP RSB 5-1 guidance that would require MPS3 to have a 36-hour safety-related SG inventory makeup supply. The BTP RSB 5-1 derived functional requirement is to have sufficient safety-related SG make-up inventory to support achieving cold shutdown in a reasonable time period. The only SG inventory make-up source credited in the MPS3 SPU SGCS analysis is the safety-related Demineralized Water Storage Tank (DWST). Thus, BTP RSB 5-1 Regulatory Position A and G are satisfied in the proposed design.

MPS3 continues to satisfy the SRP 5.4.7, Section III.5 review procedure criteria because the capability to bring the reactor to RHR entry conditions within 36-hours is unaffected.

SBPB-07-0085

In Attachment 5, Section 2.5.1.2, Missile Protection, under the results subsection, the licensee states: "For plant areas containing safety-related Structures, Systems, and Components (SSCs), the SPU will not result in any changes to existing missile sources or add any new components that could become a new potential missile source. The SPU will also not result in any system configuration changes that would impact any existing missile barrier considerations." However, the licensee determined the need to increase the feedwater pump turbine speed from 4900 revolutions per minute (rpm) to 5125 rpm in order to provide adequate flow, head, and net positive suction head to support SPU conditions. A potential source of missiles is high speed rotating components. Missiles generated internally to the reactor facility may cause damage to SSCs that are necessary for the safe shutdown of the reactor or for accident mitigation or for prevention of a significant release of radioactivity. Describe how equipment necessary for safe shutdown is protected from missiles generated by failure of the feedwater pump or its turbine. Does the increased feedwater pump operating speed affect this protection?

DNC Response

The existing turbine design rating is 12,000 HP at 5250 rpm. The turbine rotors are being replaced with the new rotors for the SPU. The redesigned rotors and the turbine will retain the original rating of 12,000 HP at 5250 rpm. The feedwater turbine SPU horsepower and operating speed of 5125 rpm are within the original feedwater turbine design rating.

In addition, the feedwater turbines are located in the turbine building near the main turbine. There is no safety related equipment required for safe reactor shutdown located in the vicinity of the feedwater pump turbines as indicated by MPS3 FSAR Table 3.5-1.

SBPB-07-0086

In Attachment 5, Section 2.5.1, Pipe Failures, the licensee addresses impact from main steam line break, recirculation pump component cooling water piping, and flooding from the high energy line break (HELB) of an SG blowdown system line in the main steam valve building. It mentions that main feedwater lines go through this area. However, the evaluation does not specifically address the increased mass release from a HELB in the feedwater system and its effect upon internal flooding. Explain the effects of increased feedwater flow from a feedwater break at SPU conditions upon internal flooding.

DNC Response

The feedwater line in the MSVB from the containment penetration to column F wall (F-Wall) is a break exclusion zone (MPS3 FSAR section 3.6.1.3.3). Therefore, a break is not postulated in this location so there is no flooding concern. From the F-Wall to the turbine building a break is postulated. There is no area for flood accumulation between the F-Wall and the Turbine Building since the ground floor is an open truck bay and the upper floors are all grating. Any discharge would quickly leave the building.

SBPB-07-0087

In Attachment 5, Section 2.5.5, Table 2.5.5.1-1 describes the changes in the operating conditions in the main steam system from current operating conditions to SPU conditions. Provide an evaluation of the change in pressure and setpoints from the high pressure turbine first stage pressure to reactor protection.

DNC Response

A new heat balance has been developed for MPS3 based upon the new SPU 100% power level, NSSS 3666 MWt. The heat balance was run for a Tavg temperature of 587.1°F and predicts a new turbine first stage pressure of approximately 712.2 psia for full load. The turbine first stage pressure transmitters will be rescaled such that the transmitter output value for full load impulse chamber pressure measurement will be changed from its present scaled full load value of 650 psia (actual plant seasonal variations 645 to 648 psia) to a new scaled full load value of 715 psia.

Because the transmitters will be rescaled to align with the new predicted full load impulse chamber pressure there will be no impact on any protection or control setpoints that use the transmitters input (Reference LAR Section 2.4.1.2.3.6 for list of functions), essentially the transmitter output for the existing full load pressure will be rescaled to be the same output for the SPU full load pressure.

During the MPS3 SPU power ascension, impulse chamber pressure will be monitored for consistency with the predicted heat balance value. If at the new 100% SPU power level the impulse chamber pressure deviates from the predicted value, the deviation will be evaluated and if necessary the turbine first stage pressure transmitters will be rescaled to accommodate the final observed value.