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December 17, 2007

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

Serial No.: 07-0799
NLOS/MAE: R1
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

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 a request for additional information (RAI) in an October 29, 2007 letter. DNC responded to the RAI in a November 19, 2007 letter (Serial No. 07-0751).

The NRC staff forwarded an additional RAI in a November 27, 2007 letter. The response to 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.

Very truly yours,

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 17th day of December, 2007.

My Commission Expires: August 31, 2008.

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

Margaret B. Bennett
Notary Public

Commitments made in this letter: Results of the updated control room fire analysis will be provided by February 29, 2008.

Attachment

cc: U.S. Nuclear Regulatory Commission
Region I
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ATTACHMENT

LICENSE AMENDMENT REQUEST

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NRC Question AFPB-07-0006

RS-001, Revision 0, *Review Standard for Extended Power Uprates*, Attachment 2 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that "power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's LAR should confirm that these elements are not impacted by the extended power uprate."

The NRC staff notes that LAR Attachment 5, Section 2.5.1.4.2.2, "Description of Analyses and Evaluations," specifically addresses only item (1) above. Provide statements to address items (2), (3), (4), and (5), and a statement confirming no increase in the potential for a radiological release resulting from a fire.

DNC Response

Stretch Power Uprate (SPU) does not affect the design or operation of fire suppression/detection systems. SPU has no impact upon fire barriers installed to satisfy NRC fire protection requirements. SPU does not affect fire protection responsibilities of plant personnel. SPU does not affect procedures and resources for the repair of systems required to achieve and maintain cold shutdown.

Any fire barrier or combustible loading changes as a result of physical modifications necessary to implement SPU will be evaluated under the licensee's fire protection program.

DNC has confirmed that there is no significant increase in the potential for a radiological release resulting from a fire.

SPU License Amendment Request (LAR) Attachment 5, Licensing Report (LR), Section 2.5.1.4.2.2, "Fire Protection, Technical Evaluation, Description of Analysis and Evaluations" addresses: (1) Administration; (2) Plant Design Features; (3) Fire Hazard Analysis; (4) Safety Shutdown Evaluation; (4) Support Systems; (5) Resolution of Safety Shutdown Evaluation Problem Areas; and

(6) Operator Action Required Following a Fire. This LR section structure mirrors the Millstone-3 Fire Protection Evaluation Report (FPER) structure.

LR Section 2.5.1.4.2.2 refers to Section 2.5.1.4.2.3 "Fire Protection, Technical Evaluation, Results" for the fire protection assessment details.

Additional Information

LR Section 2.5.1.4.2.3.2, "Fire Protection, Results, Plant Design Features" addresses portable, fixed fire suppression systems and detection. LR Section 2.5.1.4.2.3.1, "Fire Protection, Results, Administration" addresses fire protection responsibilities of plant personnel. LR Section 2.5.1.4.2.3.7, "Fire Protection, Results, Operator Actions Required Following a Fire" addresses procedures and resources for the repair of systems required to achieve and maintain cold shutdown.

LR Section 2.5.1.4.2.3.8, "Fire Protection, Results, Other Supporting Analysis/Evaluations" subsection "Risk/Potential for Radiological Release Due to a Fire" states that there is no significant increase in the potential for a radiological release from a fire at SPU conditions."

The statement that SPU has no impact upon the design and operation of fire barriers is missing from LR Section 2.5.1.4.2.3.2, "Fire Protection, Results, Fire Protection, Plant Design Features" or Section 2.5.1.4.2.3.3 "Fire Protection, Results, Fire Analysis". This RAI response provides a clear statement that SPU has no impact upon fire barriers installed to satisfy NRC fire protection requirements.

NRC Question AFPB-07-0007

LAR Attachment 5, Section 2.5.1.4.2.3.4, "Safe Shutdown Evaluations," states that
"...the safe-shutdown analysis identifies fire-induced failures that affect the plant and the operator actions that can be used to compensate for these failures..."

Discuss the response time, including any assumptions, especially those of a potentially non-conservative nature, which may have been made in determining that the operator manual actions can confidently be accomplished within the available time.

DNC Response

As described in the LAR Section 2.5.1.4.2.3.4, the revised analyses performed at SPU conditions for both Control Room Fire Transient and Charging Cubicle Fire Transient confirms that SPU does not impact the required operator action times.

The assumptions for critical operator actions used in pre-SPU and post-SPU analyses remain the same for the Control Room Fire Transient. The critical operator actions assumed following the reactor trip from the control room and initiation of main steam isolation (MSI) signal from the control room are letdown isolation and charging flow restoration. Specifically, the analysis assumes 15 minutes for letdown isolation and 30 minutes for the restoration of charging flow from the event initiation. The response times for these assumed action times have been validated for current power levels and are well within the assumed operator action times. These response times are not impacted for post SPU conditions.

Note that the response time for the manual initiation of auxiliary feedwater is discussed in AFPB-07-0008. Operator initiation time of auxiliary feedwater flow is not a parameter used in the above stated analysis.

Current analysis for Charging Cubicle Fire Transient was performed using TREAT model. There are several operator actions that occur at specific times throughout the transient that are critical in maintaining the pressurizer level on scale. Specifically, the current analysis assumes letdown isolation in 5 minutes, reactor trip in 10 minutes, closure of the MSIVs in 11 minutes and 15 minutes for securing the pressurizer heaters. The revised analysis for SPU conditions was performed using NOTRUMP model. The assumed critical operator action times remain the same as the pre SPU conditions except for the closure of MSIVs. The MSIVs are assumed closed at 660 seconds in the current analysis versus 695 seconds in the revised analysis. These analysis assumptions are bounding with respect to the Fire Shutdown Procedure of Record.

NRC Question AFPB-07-0008

LAR Attachment 5, Section 2.5.1.4.2.3.7, "Operator Actions Required Following a Fire" states that "...[an] analysis was performed to determine the steam generator dryout time at the support stretch power uprate (SPU) power level; the results showed a dryout time of approximately 37 minutes. Therefore, there continues to be adequate time for the operator to manually initiate auxiliary feedwater to the steam generators (SGs) at SPU conditions..."

Discuss the response time, including any assumptions that may have been made in determining that the operator manual actions can confidently be accomplished before SG dryout.

DNC Response

The MP3 BTP 9.5-1 Compliance Report contains information on prioritization of operator actions. One of the manual actions given high priority is auxiliary feedwater (AFW) initiation to a minimum of two Steam Generators (SG's) for the fire shutdown scenario requiring control room evacuation (i.e., CB-8, 9, 11A/B fires).

In Attachment 5, Section 2.5.1.4.2.3.7 of the LAR, the steam generator dry-out time is stated as approximately 37 minutes. During validation and verification of the response to this RAI, it was identified that there was an error in the SPU license submittal. The correct steam generator dry-out time for SPU conditions is 34.67 minutes based on the cited analysis. This value is obtained from an analysis that used a very conservative reactor trip time.

MP3 BTP 9.5-1 Compliance Report (Section 6.1.1) states that AFW flow can be initiated within 10-minutes to a minimum of two steam generators. It was recently identified that the initiation of AFW flow would occur at approximately 27 minutes for the limiting fire scenario.

Even with a decrease in steam generator dry-out time and an increase in the AFW flow initiation time, based on the availability of the large capacity turbine driven AFW pump, preliminary analysis has shown that the margin between 27 minutes and 34.67 minutes is still sufficient to assure that all BTP 9.5-1 criteria will be met. In order to provide a better estimate of the steam generator dry-out time as well as provide more complete documentation that all BTP 9.5-1 criteria can be met, an update to the control room fire analysis is in progress. Results of the updated control room fire analysis will be provided to you by February 29, 2008.

NRC Question AFPB-07-0009

LAR Attachment 5, Table 2.5.1.4-1, "Fire Shutdown and Long-Term SG Inventory Makeup Required to Support the Decay Heat Removal Design Function BTP 9.5-1 Deviation Request - Section c.5.c.3 and c.5.c.5," states that, " ...the current fire shutdown design is based upon a combined DWST [demineralized water storage tank] and CST [condensate storage tank] usable inventory that allows for 38 hours of hot standby operation, followed by a 5-hour cooldown to RHR [residual heat removal] entry conditions (38 + 5 = 43-hours)..."

Dominion Nuclear Connecticut proposed an alternative fire shutdown design approach for long-term decay heat removal to SPU after reactor trip. This is based on the DWST's 334,000-gallons of water corresponding to 13-hours of SG inventory makeup under natural circulation conditions with decay heat load after SPU, and the CST with 210,000-gallons additional SG makeup. This combined DWST and CST inventory provides 33 hours of makeup water with decay heat load after SPU.

It is not clear whether the reported 43-hours represent the current (i.e., pre-SPU) requirement for long-term hot standby operation plus cooldown or the capacity available for this combination. If the latter, then the staff notes a significant reduction in the amount of time (10-hours) in the proposed fire shutdown long-term decay heat removal approach to support SPU condition after reactor trip. If this is the case, then the staff requests the licensee to discuss the impact of this reduction in time on the post-fire safe-shutdown capability in accordance with 10 CFR Part 50, Appendix R.

On the other hand (i.e., if the former), then it follows that a shorter time (i.e., less than 33-hours) is currently needed to accomplish long-term hot standby and cooldown (i.e., pre-SPU). Therefore, an increase from this current time requirement to the 33-hours value is proposed, decreasing the margin of reserve (i.e., relative to the cited 43-hour combined capacity of the DWST and CST) under SPU. The reason for this decrease in margin, as well as the impact, would need to be discussed. The staff requests that the licensee discuss if, indeed, this is the implication of the alternative approach.

DNC Response

1. General

Due to increased decay heat, SPU causes a 9 % reduction in the available steaming time for the combined Demineralized Water Storage Tank (DWST) and Condensate Storage Tank (CST) inventory.

2. Current Fire Shutdown Design, Inventory for Sensible and Decay Heat Removal

The current fire shutdown design is described in MPS3 BTP 9.5-1 Compliance Report. The RCS decay heat removal design function is supported by the following auxiliary feedwater (AFW) pump(s) suction sources:

Table 1
Existing Fire Shutdown Design

AFW Pump Suction Source	Approximate Equivalent Steaming Time (Hours)
DWST and CST	43 (38 + 5)*
Service Water System	Unlimited

* 38-hours at hot standby, followed by a 5-hour cooldown to RHR entry conditions (38/5).

In summary, 43-hours (i.e., 38/5) represents the DWST & CST available capacity based upon a historic engineering assessment circa 1985.

3. Proposed Fire Shutdown Design, Inventory for Sensible and Decay Heat Removal

In the proposed fire shutdown design, DNC is not crediting service water (seawater) as an AFW pump suction source. Table 2 depicts the proposed fire shutdown design:

Table 2
Proposed Fire Shutdown Design

AFW Pump Suction Source	**Approximate Steaming Time (Hours)**
DWST and CST	33-hours*
Other non-seawater DWST or CST refill options (defense-in-depth/risk informed insight design approach)	19-hours (≈150,000-gallons)

* 28-hours at hot standby, followed by a 5-hour cooldown to RHR entry conditions (28/5).

For MPS3 fire shutdown events, an AFW pump suction source isn't required once a Residual Heat Removal (RHR) System train is placed in-service.

The SPU assessment of the maximum RHR entry time for a fire shutdown events identified only two fire scenarios that have RHR entry time beyond 33-hours. These fire shutdown scenarios are the AB-1 north fire (this scenario involves a Reactor Plant Component Cooling Water System Pump repair evolution) and the AB-1 south fire (this scenario involves a loss of all charging event and a boration evolution using the RWST and safety injection pumps). These two fire scenarios have a RHR entry time ≤ 52-hours.

The DWST/CST combined inventory equivalent steaming time has decreased from 43-hours (38/5) to 33-hours (28/5) which is a 10-hour reduction (or a 25% reduction). There is a 9 % steaming time reduction (4-hours) due to increased SPU decay heat. The remaining steaming time reduction is due to an increase in the CST & DWST unusable inventory allowances and a more conservative initial CST inventory assumption.

Specifically, a 50,000-gallon CST unusable volume allowance is used; consistent with the technical bases technical specification 4.7.1.3.2 "Demineralized Water Storage Tank". A 50,000-gallon condensate volume corresponds to approximately 6-hours of additional steaming time. The SPU assessment also uses a 20,000-gallon unusable DWST inventory allowance, consistent with calculations that support the technical bases for technical specification 3.7.1.3. The 50,000-gallon CST and 20,000-gallon DWST unusable volume allowances are much larger than assumed in the current fire shutdown design. In addition, an initial measured CST inventory of ≥ 210,000 gallons was used, which is conservative relative to normal CST inventory levels.

4. Impact Upon Fire Shutdown Capability

There is no adverse impact upon fire shutdown capability based upon risk informed insights because 33-hours after reactor shutdown provides ample time for DWST/CST replenishment from the available options.

The SPU licensing submittal Table 2.5.1.4-1 "Justification" Section states:

> "*The proposed fire shutdown change improves the reliability of a fission product barrier (i.e., steam generator tube integrity). Relative to the reliability of the decay heat removal design function during a fire event, there is negligible impact on the risk of radiological releases to the environment due to a fire*".

The SPU licensing submittal Attachment 1 page (26 & 27) and (pages 51 & 52) also repeats the above justification. The Risk Evaluation Section (Page 2.13-64) addresses AFW pump suction source long-term replenishment evolutions and concludes operator action time window for tank replenishment activities has not changed enough to cause a significant change in the reliability of secondary cooling design function. Section 2.13.2.3.1, "Fire Risk" contains a statement that SPU has a negligible impact on the mitigation of fires and resulting CDF due to a loss of safety functions.

Given the diverse DWST & CST refill options available (some of which are identified in Table 2.5.1.4-1), there is little risk that plant operators would fail to replenish the DWST/CST, if additional SG steaming was required beyond 33-hours.