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Dominion®

July 10, 2008

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Rockville, MD 20852-2378

Serial No.: 08-0369
NLOS/GAW: R0
Docket No.: 50-423
License No.: NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST
DNC COMMENTS ON DRAFT SAFETY EVALUATION – STRETCH POWER UPRATE

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), December 13, 2007 (Serial No. 07-0450C), March 5, 2008 (Serial No. 07-0450D), March 27, 2008 (Serial No. 07-0450E), April 24, 2008 (Serial No. 07-0450F), May 20, 2008 (Serial No. 07-0450H) and May 21, 2008 (Serial No. 07-0450I). The NRC staff forwarded requests for additional information (RAIs) in October 29, 2007, November 26, 2007, December 14, 2007, December 20, 2007 and April 23, 2008 letters. DNC responded to the RAIs in letters dated November 19, 2007 (Serial No. 07-0751), December 17, 2007 (Serial No. 07-0799), January 10, 2008 (Serial Nos. 07-0834, 07-0834A, 07-0834C, and 07-0834F), January 11, 2008 (Serial Nos. 07-0834B, 07-0834E, 07-0834G, and 07-0834H), January 14, 2008 (Serial No. 07-0834D), January 18, 2008 (Serial Nos. 07-0846, 07-0846A, 07-0846B, 07-0846C, and 07-0846D), January 31, 2008 (Serial No. 07-0834I), February 25, 2008 (Serial Nos. 07-0799A and 07-0834J), March 10, 2008 (Serial Nos. 07-0846E and 07-0846F), March 25, 2008 (Serial No. 07-0834K), April 4, 2008 (Serial No. 07-0834L), April 29, 2008 (Serial No. 08-0248), and May 15, 2008 (Serial No. 08-0248A).

On June 12, 2008, the NRC issued for comment a draft safety evaluation report regarding the stretch power uprate license amendment request. The attachment contains DNC comments on the draft safety evaluation report.

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 Mr. Geoffrey Wertz at 804-273-3572.

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 10th day of July, 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
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ATTACHMENT

LICENSE AMENDMENT REQUEST

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

DNC COMMENTS ON DRAFT SAFETY EVALUATION REPORT

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

Table of Contents

DNC Comments on Draft Safety Evaluation Report for the Stretch Power Uprate License Amendment Request

Section 1.4, Plant Modifications

Page	Location	Sentence	Comment
3	Plant Modifications, Item 4	"4. For the turbine generator, provide the following: (a) new operating point for generator excitation; (b) control valve position demand against lift settings for the valve position cards; (c) changes to load imbalance circuits; (d) throttle pressure and excess throttle pressure circuit recalibrations; (e) sensor rescaling for steam pressure changes; (f) instrument scaling; and (g) main control board and panel meter replacements."	Please delete "(a) new operating point for generator excitation", "(d) throttle pressure and excess throttle pressure circuit recalibrations" and "(g) main control board and panel meter replacements." Please refer to DNC letter dated May 20, 2008 (Serial No. 07-0450H), Attachment, Updated Table 1.0-1 MPS3 Power Uprate Planned Modifications, Page 2, "Turbine Generator" which revised the list of planned modifications.
4	Plant Modifications, Item 7	"7. Pipe support modifications for the condensate system, feedwater system, CCW system, and extraction steam system."	Please replace "extraction steam" with "containment recirculation." Please refer to DNC letter dated May 20, 2008 (Serial No. 07-0450H), Attachment, Updated Table 1.0-1 MPS3 Power Uprate Planned Modifications, Page 3, "Pipe Support Modifications, Condensate, Feedwater, Component Cooling water, and Containment Recirculation," which revised the list of planned modifications.

Page	Location	Sentence	Comment
4	Plant Modifications, Item 9	"9. Provide instrument loop rescaling for the following: (a) isophase bus duct cooler flow; (b) moisture separator reheater steam flow; and (c) first stage turbine pressure."	Please delete "(a) isophase bus duct cooler flow" and "(b) moisture separator reheater steam flow." Please refer to DNC letter dated May 20, 2008 (Serial No. 07-0450H), Attachment, Updated Table 1.0-1 MPS3 Power Uprate Planned Modifications, Page 3, "Instrument Loop Rescaling" which revised the list of planned modifications.

Section 2.1, Materials and Chemical Engineering

Page	Location	Sentence	Comment
8	2nd paragraph, last sentence	"The methodology used to project the neutron fluence values for surveillance capsules X (WCAP-15045) and W (WCAP-16629-Reference 2) adhered to the guidance in Regulatory Guide (RG) 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."	As stated in DNC letter dated November 19, 2007 (Serial No. 07-0751), in response to RAI CVIB-07-002, the fluence analyses per WCAP-15405 were compliant with the requirements of Draft Regulatory Guide DG-1053, which was the precursor to Regulatory Guide 1.190. A more accurate statement would be: "The methodology used to project the neutron fluence values for surveillance capsules X (WCAP-15405) and W (WCAP-16629-Reference 2) adhered to the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," or its predecessor."
8	3rd paragraph, first sentence	"The projected peak neutron fluence values on the inside surface of the RV for 32 and 54 EFPYs are 1.97 x 10^{19} n/cm^2 (E > 1 MeV) and 3.31 x 10^{19} n/cm2 (E > 1 MeV) respectively. In addition, the calculation assumed that the equilibrium cycle 15 core loading will be implemented through the 54 EFPYs."	As stated in DNC letter dated November 19, 2007 (Serial No. 07-0751) in response to RAI CVIB 07-003, the fluences given in these sentences correspond to earlier projections based on capsule X with the assumption that the flux values from cycle 4 through 6 would continue throughout plant life. While these values are conservative, the values more appropriate to SPU are 1.63 x 10^{19} n/cm^2 and 2.70 x 10^{19} n/cm^2 for 32 EFPY and 54 EFPY respectively.

Page	Location	Sentence	Comment
11	Last paragraph, third sentence	"The licensee used position 2.1 of the RG 1.99 Revision 2, where credible surveillance capsule test data was used for establishing RT_{PTS} value for the intermediate shell plate."	It should be noted that information is provided in Table 2.1.3-4 of the License Report for both Position 1.1 and Position 2.1. As stated on page 2.1-35 of the License Report, "The limiting material is Intermediate Shell Plate B9805-1, with the more limiting RT_{PTS} value occurring for calculations using the RG 1.99, Rev. 2 Position 1.1 Chemistry as opposed to the Position 2.1 Chemistry Factor calculated from credible surveillance data."
12	Second paragraph, last sentence	"Specific review criteria are contained in SRP Section 4.5.2, WCAP-14277,[21] and BAW-2248.[22]"	Please replace "WCAP-14277,[21]" with "WCAP-14577,[21]". WCAP-14577 is the correct reference for the License Renewal Evaluations: Aging Management for Reactor Internals.
22	Flow-Accelerated Corrosion, First paragraph, sixth sentence	"The licensee's FAC program is based on NUREG-1344, "Embrittlement Criteria for Zircaloy Fuel Cladding Applicable to Accident Situations in Light-Water Reactors: Summary Report, ..."	The statement should read "The licensee's FAC program is based on NRC Bulletin 87-01, "Thinning Pipe Walls in Nuclear Power Plants", ..." This is stated in the licensing report, page 2.1-76 under MPS3 Current Licensing Bases.

Section 2.2, Mechanical and Civil Engineering

Page	Location	Sentence	Comment
34	Second paragraph, First line under "Reactor Vessel and Supports"	"The licensee performed its evaluations for the MPS3 RPV and its supports at SPU conditions in accordance with the current plant code of record, ASME Code, Section III, Division 1, 1971 Edition through summer 1973."	The wording "and its supports" should be removed since the code of record is different for the RPV Supports. The RPV supports are ASME III 1974 Edition including Summer 74 addenda. Note that in this paragraph, the reference is made to FSAR Table 5.4-18 which provides the correct information. Also please add the word "addenda" after "summer 1973."
35	First paragraph, third sentence	"In its response to the NRC staff's RAI, the licensee provided a summary of the results of the elastic-plastic evaluation which shows that the special rules for exceeding 3Sm as provided by (a) through (f) of Subparagraph NB-3228.3 have been met."	Please use "simplified elastic-plastic evaluation" instead of "elastic-plastic evaluation." This is stated in DNC letter dated April 4, 2008, (Serial No. 07-0834L) page 1 of the Attachment.
36	Last paragraph, seventh line.	"The Code of Record for the SG supports is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF 1974 Edition including 1974 summer Addenda."	The word: "summer" should be "winter." The correct reference is provided in FSAR Table 5.4-18, "Equipment Supports, Loading Combinations, and Design Allowable Stresses."
39	End of First Paragraph at top of page.	"...Section III, Division 1, 1974 Edition through summer 1974 Addenda."	The word: "summer" should be "winter." The correct reference is provided in FSAR Table 5.4-18.
44	Safety-Related Valves and Pumps, second paragraph,	"The licensee has reviewed system level design basis calculations for main steam system (MSS), service water system, and	Please insert "Cat 1 AOVs in the" between "for" and "main steam." The following is a quotation from the license amendment report

Page	Location	Sentence	Comment
	second sentence	CVCS system."	section 2.2.4, page 2.2-141: "The system level design basis review calculations for the Category 1 AOVs in the following systems were reviewed: • Main steam system (main steam pressure relieving valves, turbine-driven auxiliary feedwater pump steam supply valves) • Service water system (service water diesel generator heat exchanger outlet valves) • Chemical and volume control system (reactor coolant letdown inside/outside Containment isolation valves)"
44	Safety-Related Valves and Pumps, fourth paragraph, second sentence	"The Code of record for MPS3 is the 1980 Edition through winter 1980 Addenda of the ASME Code, Section XI."	Please insert "original" between "The" and "Code". As stated in the last sentence of the paragraph, "The IST program must be periodically updated to meet applicable ASME O&M Code Requirements specified in 10 CFR 50.55a," which means that the code years and addenda are updated as required. The "original" code was 1980 Edition through winter 1980 addenda – This would be a clarification.

Section 2.3, Electrical Engineering

Page	Location	Sentence	Comment
49	Main Steam Valve Building, third paragraph, fourth sentence from the bottom	"The valves fail closed, and the failure of the limit switches would not cause the reopening of the valve."	Please replace with "Once the valves move to their fail safe position, closed, the failure of the limit switches would not cause the reopening of the valve." Please refer to the response to question EEEB-07-0052 Follow-Up (EEEB-08-0108) in DNC letter dated May 15, 2008 (Serial No. 08-0248A).
50	Turbine Building, third line of the last paragraph	"The pressure for the turbine building remains bounding for SPU conditions."	Please add "from a MSLB" after "turbine building." The phrase "from a MSLB" is missing.
51	Turbine Building, second sentence in the second paragraph	"The primary function of the transmitters is to provide input to the rod control system, where a transmitter failure could initiate a rod withdrawal demand signal coincident with a steam line break."	The second sentence in the second paragraph does not read the same as intended in the RAI response to questions EEEB-07-0056 and EEEB-07-0057 in DNC letter dated January 10, 2008 (Serial No. 07-0834C). Please consider replacing the sentence with: "One function of the transmitters is to provide input to the rod control system, where a transmitter failure could initiate a rod withdrawal demand signal coincident with a steam line break."

Section 2.5, Plant Systems

Page	Location	Sentence	Comment
61	Section 2.5.1.1.1 Flooding Protection, Technical Evaluation, first Paragraph, third line	" ..... and vessels was based on instantaneous release of fluid from the tanks."	Please delete the word "instantaneous" since there is no basis for the use of instantaneous. The license amendment report does not discuss "instantaneous release."
63	Technical Evaluation, First line of the first paragraph	"As discussed in UFSAR Section 3.8.4, the CWS provides a continuous supply of cooling water to the MC to remove excess heat from the steam turbine exhaust cycle and auxiliary systems."	UFSAR Section 3.8.4 does not discuss circulating water system heat removal. Therefore please delete the phrase "As discussed in UFSAR Section 3.8.4."
63	Conclusion, third line	"the increased volumes of fluid leakage that could potentially result from these modifications would not result in the failure of safety-related SSCs following implementation of the proposed SPU."	This section reads as though DNC is performing modifications to the CWS and that there is an increase in fluid leakage volume. Please consider revising the conclusion as suggested below: "Since the CWS flow and operating pressures will remain unchanged for the SPU, and no modifications to the CWS or turbine building are performed that would affect the analyses associated with flooding, the proposed SPU is acceptable with respect to flooding of CWS and would not result in the failure of safety-related SSCs following implementation of the proposed SPU."
66	Second paragraph	"With respect to a postulated MSLB that results in an increase in the turbine building peak temperature from 500 °F to 565.5 °F, the previously performed	This paragraph has mixed up the results for the turbine building and the auxiliary building with the results for the main steam valve building.

Page	Location	Sentence	Comment
		temperature analysis is no longer bounding for SPU conditions. The impact on the qualification of equipment as a result of an MSLB, is discussed in Section 2.3 of this report. The increase in peak temperature may also impact specific valves that are essential for shutdown in the event of an HELB in the auxiliary building. MPS3 UFSAR Table 3.6-5 lists the two specific valves. The EQ of these two valves to remain effective under increased temperature condition is also discussed in Section 2.3 of this report."	The LAR section for a TB HELB is located on LAR page 2.5-28 (bottom). This LAR says that DNC is not evaluating the TB HELB temperature increase because DNC will change the TB HELB safe shutdown analysis to eliminate crediting any equipment located in the TB. There are two isolation valves located in the TB that are credited for an AB HELB (i.e., 3ASS-AOV102A/B). The subject SER paragraph discusses these valves. SPU has no impact upon the AB temperature/pressure profile associated with an AB HELB. Because the AB temperature/pressure is unaffected by SPU and the AB is not adjacent to the TB, an AB HELB has no impact upon TB temperature/pressure. Thus, SPU has no impact upon the AB HELB credited steam isolation valves (i.e., 3ASS-AOV102A/B) that are located in the TB. The "500 °F to 565.5 °F" temperature increase is associated with the Main Steam Valve Building (MSVB). Specifically, this temperature increase is for a MSVB MSLB event. DNC suggests simplifying/clarifying the

Page	Location	Sentence	Comment
			subject paragraph as follows: "For a Main Steam Valve Building (MSVB) HELB, MSVB temperature qualification profiles are impacted (i.e., peak temperature increases from 500 °F to 565.5 °F). Section 2.3 documents assessment of MSVB harsh environmental changes with respect to credited SSC's. For a Turbine Building (TB) HELB, SPU will increase the TB HELB temperature. However, to address this consequence, the licensee proposes to revise the TB HELB safe shutdown analysis to eliminate crediting equipment located in the TB. In the proposed design, a TB HELB has no adverse impact on credited SSC's because no credited equipment is located in the TB and there is no impact to adjacent areas due to building layout/design. Review of pipe failure assessments for all other buildings concluded that SPU has no adverse impact upon existing moderate or high energy pipe failure analysis."

Page	Location	Sentence	Comment
69	First paragraph, Last sentence	"The licensee indicated that for the proposed SPU condition, there is no increase in the potential for a radiological release resulting from a fire."	Consistent with RAI response to Question AFPB-07-0006, please consider revising from "there is no increase in the.." to "there is no significant increase in the ..."
69	Sections after the first two paragraphs	"By letter dated December 17, 2007."	A more up to date response to RAI AFPB-07-0007 is provided in letter dated February 25, 2008. The response in the letter dated December 17 2007 was superseded by the response in the letter dated February 25, 2008. The primary difference between the two responses is the clarification of the MSIV closure time in the later response. Therefore please use the response from February 25, 2008 letter.
71	Second paragraph, third line	"The licensee also stated that in order to provide margin, the AFW initiation time will be reduced to less than 21.5 minutes prior to implementation of the SPU by making changes to the fire shutdown and design compliance report."	Please consider changing from "fire shutdown and design compliance report" to "fire shutdown procedure and design compliance report."

Page	Location	Sentence	Comment
78	Technical Evaluation, second sentence	"No modifications are being made to the TGSS and non-condensable gases will continue to be monitored for radiation."	DNC proposes to replace this sentence with: "No physical changes to system components or changes in system operation are required due to the slight increase in sealing flow and gland steam condenser cooling flow. There is no radiation monitoring at the gland seal condenser vent, as radioactive gaseous releases fall within the total unmonitored steam release specifications from the Turbine Building as defined in NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors, April 1976, Section 2.2.6." The proposed change reflects the results of DNC evaluation as provided on pages 2.5-64 and 2.5-66 of the licensing report.
81	First sentence, last paragraph	"The operating band of the CCW surge tank is sufficient to accept a 100 gallon increase in system thermal expansion volume relative to the current system volume."	Since the operating band of the CCW surge tank can accept more than 100 gallons, please revise the sentence to read as "The operating band of the CCW surge tank is sufficient to accept the increase in system thermal expansion volume relative to the current volume." The LAR section 2.5.4.3.2.3 describes an exact value of 100 gallons which is an estimated value.

Page	Location	Sentence	Comment
84	Fourth paragraph, second sentence	"Upon a loss of normal feedwater (LONF), the licensee's proposed 7-hours at hot shutdown with higher decay heat as a result of SPU does not impact the available water volume in DWST to provide cooldown."	The proposed 7/6 DWST sizing criteria is 7-hours at hot standby. Therefore, DNC suggests changing "7-hours at hot shutdown" to "7-hours at hot standby."
85	Third paragraph, first sentence	"The DWST inventory requirement is also relative to fire shutdown."	DNC suggests using: "The DWST inventory is also credited within the fire shutdown analysis."

Page	Location	Sentence	Comment
86	Technical Evaluation, first sentence	"The proposed SPU would increase steam temperature to 533 °F from 500 °F."	DNC proposes replacing the sentence with: "The proposed SPU would increase steam temperature of the reheat lines to 533 °F from 500 °F." The temperature increase to 533 °F is associated only with the hot reheat lines as discussed in section 2.5.5 of the license report, page 2.5-111, under System/Component Design Pressure/Temperature Versus SPU.
89	Technical Evaluation, first paragraph.	"The licensee's analysis of a TT without reactor trip showed the pressurizer PORVs could be challenged at SPU conditions. A proposed modification of the steam dump load rejection controller "Hi 2" trip open setpoint from 12.0 °F to 7.0 °F, prior to the start of the coast down maneuver, and reset the controller back to 12.0 °F prior to the subsequent startup for the new fuel cycle at SPU conditions would ensure that a TT would not result in a challenge to the pressurizer relief valve lift setpoint. During a load runback transient, the MSS pressure would remain less than the MSSV setpoint."	Please replace with: "The licensee's analysis of a TT without reactor trip showed the pressurizer PORVs could be challenged at SPU conditions. A proposed modification to the steam dump load rejection controller will include a change to both the Hi 1 and Hi 2 setpoints. Hi 1 will be changed from 6.6 °F to 5.3 °F. Hi 2 will be changed from 15.8 °F to 12.0 °F. Should the plant choose to use the alternate coastdown method identified as Option 2 in license amendment report Table 2.4.1-4, then the Hi 2 setpoint will be changed to 7.0 °F for the coasdown and then reset to the new at power Hi 2 setpoint of 12.0 °F prior to subsequent startup for the new fuel cycle. Otherwise the Hi 2 setpoint will remain at 12.0 °F for the Option 1 coastdown. The modified setpoints would ensure that a TT would not result in a challenge to the pressurizer relief valve lift

			setpoint. During a load runback transient, the MSS pressure would remain less than the MSSV setpoint." Please refer to table 2.4.1-4 (page 2.4-24) of the license amendment report.

Section 2.6, Containment Review Considerations

Page	Location	Sentence	Comment
97	Regulatory Evaluation 1st paragraph, last sentence	"The containment maximum internal design pressure is 45 pounds per square inch gauge (psig), the minimum internal design pressure is 8.00 psia, and the design temperature is 296°F."	MPS3 does not apply a maximum containment vapor design temperature limit of 296°F. The containment structural integrity is assured by applying a containment liner temperature limit of 280°F. For equipment inside containment, a temperature qualification profile is used for qualification of equipment inside containment. As can seen from LR section 2.6.1.2.3.3.2, page 2.6-14, the predicted short term peak from a steam line break is 343°F which exceeds 296°F. This is also the case for the current licensing basis results documented in FSAR Section 6.2.1.1.3.7, page 6.2-16 where the predicted short term peak from a steam line break is 335.9°F. Please consider revising or deleting this sentence.
98	LOCA Short-Term Containment Response last sentence	"The LOCA short-term peak pressure and temperature are less than the containment design pressure and temperature 45 psig and 296°F respectively."	MPS3 does not apply a maximum containment vapor design temperature limit of 296°F. See discussion above. Please consider revising or deleting this sentence.
98	LOCA Long-Term Containment Response 2nd sentence.	"The initial conditions of maximum pressure, maximum temperature and minimum relative humidity along with a single failure or loss of an EDG produces the slowest containment depressurization for this break."	This should read "The initial conditions of maximum pressure, maximum temperature and maximum relative humidity…" This is stated on page 2.6-11 of the LR.

Page	Location	Sentence	Comment
99	Main Steam Line Break Containment Response 1st paragraph, third sentence	"The peak pressure occurs at 12.6 seconds from the time of the instantaneous break."	The 12.6 seconds should be 194.3 seconds. This is stated in Table 2.6.1.2.3-2 of the LR.

Section 2.7, Habitability, Filtration, and Ventilation

Page	Location	Sentence	Comment
108	Technical Evaluation, First paragraph, item (2)	"(2) for a fuel handling accident (FHA), the CR emergency ventilation system (CREVS) is required to be in the pressurized filtration MODE within 30 minutes of the accident, whereas in the current licensing basis analysis it is required to be in the pressurized filtration MODE within 1.685 hours[97] of the accident."	Please consider changing "pressurized filtration" to "filtered recirculation." This change will make this item consistent with other sections of the SER. For example, the term 'filtered recirculation' is used for the condition of the control room 30 minutes post FHA. See section 2.9.2.2, page 199, 2nd, 3rd and last paragraph; Control Room Habitability, page 211, 1st and 4th paragraph; Changes in Emergency and Abnormal Operating procedures, page 229, 1st bullet.

Section 2.8, Reactor Systems

As part of DNC review of the NRC's draft SPU safety evaluation report (SER), errors were identified in Table 2.8.2-1, "Range of Key Safety Parameters Safety Parameter," page 2.8-18 of attachment 5 of the license amendment request dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A). The value given as the current vessel average temperature is the core average temperature rather than the vessel average temperature. Footnote c was incorrectly assigned to "Most Positive MDC (K/g/cm3)." The following are the proposed corrections:

- The current vessel average temperature is 587.1 °F.
- Foot note c "Linear ramp from 70% to 100% power" applies to the MTC >70%.
- Footnote d applies to the second entry for the Most Positive MDC.
- Footnote d "MDC assumed for feedline break analysis (HFP ARO)" applies to the Most Positive MDC of 0.45.

The values used in Table Line item 6 and 7, on Page 120 of the SER, came from license report Table 2.8.2-1. Therefore, DNC proposes changes to the following two lines in order to incorporate the above mentioned corrections.

Replace the following three lines:

	Current Design	Uprate Analysis Values
Vessel Average Coolant Temp. HFP (°F)	591.6	581.5 to 589.5 [a,b]
Most Positive MTC, >70% (pcm/°F)	0.0	0.0
Most Positive MCC ***	0.50	0.50, linear ramp from 70% to 100% power to 0.45

With:

	Current Design	Uprate Analysis Values
Vessel Average Coolant Temp. HFP (°F)	587.1	581.5 to 589.5 [a,b]
Most Positive MTC, >70% (pcm/°F)	0.0, linear ramp from 70% to 100% power	0.0 , linear ramp from 70% to 100% power
Most Positive MDC	0.50	0.50 0.45 for FWLB at HFP ARO

The following is the corrected license report Table 2.8.2-1:

Range of Key Safety Parameters Safety Parameter

	Current Design Values	Analysis Values
Reactor Core Power (MWt)	3411	3650
Vessel Average Coolant Temp. HFP (°F)	587.1	581.5 to 589.5 [a,b]
Coolant System Pressure (psia)	2250	2250
Core Average Linear Heat Rate (kW/ft)	5.45	5.83
Most Positive MTC (pcm/°F)		
Power < 70%	+ 5.0	+ 5.0
Power ≥ 70%	0.0[c]	0.0[c]
Most Positive MDC (K/g/cm3)	0.50	0.50
		0.45[d]
Doppler Temperature Coefficient (pcm/°F)	-3.20 to -0.91	-3.20 to -0.90
Doppler Only Power Coefficient (pcm/%Power)		
Least Negative, 118%RTP to HZP	-9.55 to -5.42	----
Most Negative, 118%RTP to HZP	-19.40 to -11.36	----
Least Negative, 121%RTP to HZP	-11.55 to -7.02	----
Most Negative, 121%RTP to HZP	-19.40 to -11.17	----
Least Negative, 130%RTP to HZP	----	-9.55 to -5.00
Most Negative, 130%RTP to HZP	----	-19.40 to -10.56
Beta-Effective	0.0040 to 0.0070	0.0040 to 0.0075

	Current Design Values	Analysis Values
Normal Operation FNH	1.70	1.65
Normal Operation FQ(Z)	2.60	2.60
Shutdown Margin (%)	1.30	1.30

a. The vessel average coolant temperature can decrease to 571.5 ° F during a coast down.
b. Constant temperature program assumed during nominal depletion.
c. Linear ramp from 70% to 100% power.
d. MDC assumed for feedline break analysis (HFP ARO)

Section 2.8, Reactor Systems (continue)

Page	Location	Sentence	Comment
116	Technical Evaluation 1st paragraph	"The fuel system in use at MPS3 includes Westinghouse 17X17fuel matrices of the Standard (STD), VANTAGE 5 Hybrid (V5H), Robust Fuel Assembly (RFA) and RFA-2 designs."	While these fuel types have been used in the past, for the SPU cycle only RFA-2 assemblies will be in use. A more accurate sentence would be "The fuel systems that have been placed in service at MPS3 include Westinghouse 17x17..."
117	5th paragraph	"Similar to evaluation at the current license conditions, fuel rod performance for the uprated core was evaluated using a reference fuel system comprised entirely of RFA fuel..."	For accuracy, this should read "... reference fuel system comprised entirely of RFA/RFA-2 fuel..." In fact, the first SPU uprate cycle will consist of entirely RFA-2 assemblies.

Page	Location	Sentence	Comment
129	Last sentence	"The licensee responded by letter dated January 11, 2008, stating that the peak shell side design temperature of the RHRS heat exchanger is 200 °F, which is well over the newly proposed CCW return flow design temperature of 145 °F."	DNC suggests changing the part: "proposed CCW return flow design temperature of 145°F" to: "proposed CCW return line 145°F operating temperature." The LAR Table 2.8.4.4-2, "Proposed Cooldown Related Design Changes" explains that 145°F is a maximum operating temperature. The design temperature (and stress analyzed temperature) for the subject CCW lines is increased from 150°F to 160°F.
130	Second paragraph, second sentence.	"The licensee stated that, for normal cooldown, this evolution will take place within 4-hours."	DNC suggests changing the sentence to read: "The licensee stated that for the normal cooldown analysis, RHR entry occurs at 4-hours after a reactor trip." The proposed change reflects the information provided in LAR Section 2.8.4.4.2.2.1.
130	Sixth paragraph, first sentence	"Because there was no change in cooldown time for normal RHRS cooldown, the NRC staff requested additional information regarding differences in the normal and safety-grade cold shutdown analyses from pre- to post-uprate implementation."	DNC suggests changing the part: "Because there is no change in cooldown time for normal RHRS cooldown ..." to: "Because there is no increase in cooldown time for normal RHRS cooldown ..." Per LAR Table 2.8.4.4-1, there is a decrease in normal cooldown times due to SPU and the associated SPU design changes.

Page	Location	Sentence	Comment
133	1st paragraph, fourth sentence	"However, the NRC staff does not consider the reactor trip to be the direct mitigation for this increase in feedwater flow event, especially since the minimum DNBR occurs only 0.3 seconds after rod motion is begun."	From License Report page 2.8-125, Table 2.8.5.1.1.2.2-2, the sequence of events shows the following: Reactor Trip on Turbine Trip – rod motion initiated: 32.4 seconds Minimum DNBR (1.88) reached: 32.5 seconds. The difference is 0.1 seconds rather than 0.3 seconds.
133	Second paragraph, last sentence	"Therefore, the NRC staff finds these results to be acceptable, despite the crediting of reactor trip on TT in the analysis."	DNC fully concurs with the NRC that credit cannot be taken for the reactor trip from the Turbine Trip signal. As correctly discussed by the NRC, once feedwater isolation occurs, the reactivity addition and power rise will be terminated and the RCS will begin to pressurize. This will result in an increase in DNBR. The reactor trip following turbine trip was modeled to terminate the transient analysis at a point beyond the time of minimum DNBR. As such the reactor trip does not represent a credit in determining the minimum DNBR. A more accurate wording is : "Therefore, the NRC staff finds these results to be acceptable, despite the modeling of reactor trip on TT in the analysis."

Page	Location	Sentence	Comment
135	4th paragraph in Technical Evaluation, second sentence	"However, the sequence of events (LR Table 2.8.5.1.2.2.1-1) lists low steam line pressure as the first safety injection signal."	Although the low steam line pressure setpoint is reached first, and it generates an SI signal (which initiates feedwater isolation), the analysis does not credit the initiation of SI flow until after the SI signal from low pressurizer pressure is generated at 25.8 seconds. This timing is reflected in the following comment.
135	5th paragraph in Technical Evaluation, first sentence	"Although the safety injection system is actuated early in the transient (0.5 seconds) the minimum ..."	Per LR Table 2.8.5.1.2.2.1-1, the SI flow starts at 72.8 seconds.
136	5th paragraph	"The limiting case... which is less than the peak linear heat rate SAL of ..."	Westinghouse considers the peak linear heat rate SAL as proprietary. The sentence can be re-worded as follows to eliminate the proprietary information. "The limiting case...which is less than the peak linear heat rate SAL given in LR Table 2.8.3-1.

Page	Location	Sentence	Comment
140	1st paragraph and items 1 through 3	"Analyses for the loss-of-non-emergency ac power event are not reported in the LR, since this event is bounded by the complete loss-of-flow event: 1. LR Section 2.8.5.3, with respect to the DNBR SAL, by the LOL/TT event; 2. LR Section 2.8.5.2.1, with respect to RCS pressure and MSS pressure SALs, and by the LONF event with loss-of-non-emergency ac power; 3. LR Section 2.8.5.2.3, with respect to the capabilities of RCS natural circulation and the AFWS to remove stored and residual heat."	The following rewording is recommended: "Analyses for the loss-of-non-emergency ac power event are not reported in the LR, since this event is bounded by: 1. the complete loss of flow event, LR Section 2.8.5.3 with respect to DNBR SAL; 2. the LOL/TT event, LR Section 2.8.5.2.1 with respect to RCS pressure and MSS pressure SALs and 3. the LONF event with loss of non-emergency ac power, LR Section 2.8.5.2.3, with respect to the capabilities of RCS natural circulation and the AFWS to remove stored and residual heat."
140	Conclusion	"The NRC staff has reviewed the licensee's analyses of the loss-of-non-emergency ac power to station auxiliaries event...."	This sentence is inconsistent with the first sentence on this page which states "Analyses for the loss-of-non-emergency ac power event are not reported in the LR.." The following rewording is recommended: "The NRC staff has reviewed the licensee's discussion of the loss-of-non-emergency..."
141	Technical Evaluation 1st paragraph, last sentence	"The AFW system starts automatically on SG low-low water level, following a safety injection signal, on LOOP, or on trip of all main feedwater pumps."	While this is an accurate statement, it is noted that only the two motor driven AFW pumps will start automatically following a safety injection signal, loss of offsite power or trip of the Main feedwater pumps. Only the SG low-low level signal will automatically start all pumps, the

Page	Location	Sentence	Comment
			turbine driven AFW pump as well as the two motor driven AFW pumps. It is recommended that this sentence be clarified to avoid confusion in the future.
140	Last paragraph, third sentence	"RCP heats of 20 MWt and 16 MWt were added to the cases with and without offsite power."	A more accurate statement would be "RCP heats of 20 MWt and 16 MWt were modeled in the cases with and without offsite power" as is stated in License Report 2.8-183. Post-trip, for the offsite power cases, 20 MWt was included as heat source from the RCPs. Post-trip for the loss of offsite power cases, the RCP pumps are not a heat source since they will coastdown. The initial power level was assumed to be 1.02 times 3666 MWt, which includes 16 MWt for RCP pump heat.
144	4th full paragraph	"There are FLB analyses results, available for MPS3, for both assumptions. The licensing basis FLB analyses, in which the pressurizer PORVs are assumed to be available, predict that the minimum RCS subcooling would be 22°F."	This would be clearer if it is re-written as follows: "A comparison of the current licensing basis FLB analysis and the SPU FLB analysis can be used to evaluate this assumption. The current licensing basis FLB analyses,..." Note: While not documented in the Licensing Report, it should be noted that the SPU FLB analyses included a number of sensitivity studies including the impact of PORV availability. With credit taken for the PORVs, the margin to hot leg saturation was 15°F, demonstrating that the assumption of the PORVs being unavailable is conservative at SPU conditions.

Page	Location	Sentence	Comment
149	2nd full paragraph.	"The licensee analyzed a postulated locked rotor, and noted that the consequences of the locked rotor accident are very similar to those of an RCP shaft break."	The RETRAN model for this event took into account both the impact of the locked rotor and the sheared shaft. A locked rotor will result in fastest reduction in forward flow through the loop. A sheared shaft will result in the largest reverse flow in the affected loop, consequently, resulting in the maximum reduction in the core flow once the flow in the affected loop reverses. The RCP homologous pump curve in the forward flow direction was modified to take into account the increased resistance due to the locked rotor. The RCP homologous pump curve in the reverse flow direction was modified to take into account the reduced resistance due to a sheared shaft. This information provides additional basis for the statement in the SER.
154	Last sentence	"Therefore, the licensee concluded, the generic analysis was not applicable to MPS3, and the results of the full power analysis presented in the SPULR adequately demonstrated that the plant would not overpressurize during an RWAP transient."	A more accurate wording for this sentence is as follows: "Therefore, the licensee concluded, reference to the generic analysis was not required for MPS3, and the results of the full power analysis presented in the SPULR adequately demonstrated that the plant would not overpressurize during an RWAP transient." As stated in DNC letter 07-0450F dated April 24, 2008, DNC believes that it was unnecessary to reference the generic analysis. The generic analysis while bounding of MP3, was overly conservative.

Page	Location	Sentence	Comment
157	Conclusion 1st sentence	"The NRC staff has reviewed the licensee's analyses of control ..."	This sentence is inconsistent with the discussion at the top of this page. As stated at the top of the page, generic dropped RCCA statepoints are used rather than licensee analyses. The following re-wording is recommended: "The NRC staff has reviewed the licensee's discussion of control rod misoperation..."
158	Conclusion 1st sentence.	"The NRC staff has reviewed the licensee's analyses of the inactive loop startup event..."	This is inconsistent with the last sentence in the Technical Evaluation where it states "The staff finds that this event need not be analyzed to implement the proposed power uprate." The following re-wording for this paragraph is recommended: "The staff has reviewed the licensee's discussion of the inactive loop startup event and concludes that this event need not be analyzed to implement the proposed power uprate. Therefore, the NRC staff finds the proposed SPU acceptable with respect to the increase in core flow event."

Page	Location	Sentence	Comment
164	3rd full paragraph	"The LR indicates that, "ECCS injection is not credited for mitigating steam line or feedwater line breaks." Actually, ECCS injection is credited for mitigating steam line and FLBs."	DNC concurs that this is an inaccurate statement. However, DNC could not identify where this statement is made. The ECCS modeling for the SLB and FLB SPU analyses is as follows: For Steam Line Break DNBR analyses, no credit is taken for boron addition from the charging pumps. For Steam Line Break Mass and Energy releases, ECCS charging flow is modeled. However, ECCS flow is assumed to be initiated on low pressurizer pressure and this is unaffected by the implementation of the P-19 permissive. For FLB, charging flow ECCS is credited for assisting in maintaining hot leg saturation. However, ECCS flow is assumed to be initiated on low pressurizer pressure and thus is unaffected by the implementation of the P-19 permissive.
164	Last sentence	"The P-19 permissive essentially eliminates the ability of the ECCS to deliver flow to the RCS cold legs at pressures above the low pressurizer pressure reactor setpoint."	A more accurate statement is as follows: "The P-19 permissive essentially eliminates the ability of the ECCS to automatically deliver flow to the RCS cold legs at pressures above the low pressurizer pressure reactor trip setpoint." The sentence as written could be interpreted to mean that the operators would not be able to perform feed-and-bleed. The P-19 permissive has no impact on the operator to manually initiate ECCS charging flow to the colds legs

Page	Location	Sentence	Comment
			when required for beyond design basis events such as feed-and-bleed.
169	Top paragraph, last sentence	"Preventing SG overfill is necessary in order to prevent the failure of the steam lines."	The most significant reason to demonstrate that the SG will not overfill is to assure that credit can be taken for iodine partitioning in the SG. If water is released from the SG directly to the environment, a significant increase in iodine releases will be predicted since a DF of 1 will be assumed. In fact, the MPS3 steam lines are designed for the loads of a water solid condition. This information is contained in NU letter B12787 to the NRC dated January 22, 1988, "Millstone Nuclear Power Station, Unit No. 3 Steam Generator Tube Rupture (SGTR) – Plant Specific Information." The following re-wording is recommended: "Preventing SG overfill is necessary in order to prevent radioactive liquid releases to the environment and to prevent the failure of the steam lines."
170	4th paragraph, fifth sentence	"These actions are defined in the Westinghouse Owner's Group Emergency Response E-3 Guidelines."	A more accurate statement would be "These actions are defined in the Westinghouse Owner's Group Emergency Response E-3 Series of Guidelines." The E-3 Series consists of E-3 Steam Generator Tube Rupture; ES-3.1 Post SGTR Cooldown Using Backfill; ES-3.2 Post-SGTR Cooldown Using Blowdown and ES-3.3 Post-SGTR Cooldown Using Steam Dump.

Section 2.9, Source Terms and Radiological Consequences Analyses

Page	Location	Sentence	Comment
190	Section 2.9.2, second to the last line	"The inventory, consisting of 66 isotopes at end of fuel cycle curie levels, formed the input for the RADTRAD-NAI dose evaluation code."	Please change 66 isotopes to 72 isotopes. The basis for the number 72 is contained in SPU Licensing Report, Table 2.9.2-1, (note that the RADTRAD-NAI inventory file contains 72 isotopes, but Co-58 and Co-60 are not listed in the Table due to 0.0 curies). Also, RAI Response to question AADB-07-0107 in the letter dated 1/18/08 (Serial No. 07-0846), Attachment 3, page 4 of 106.
192	Section 2.9.2.1, second paragraph, second sentence	"For conservatism the licensee maintained the partitioning of the fission products source between the sprayed/unsprayed regions constant (0.4963/0.5037) during this period."	The values given for sprayed/unsprayed region applies only to the period when quench spray alone is in operation. Different values are used for when quench spray and recirculation spray are in operation together and when recirculation spray is in operation alone. Please change to: "The licensee maintained the partitioning of the fission product source between the sprayed/unsprayed regions consistent with the percentage of containment sprayed/unsprayed during these periods." The basis for this change is contained in DNC response to RAI question AADP-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 3, page 4 of 106.

Page	Location	Sentence	Comment
194	Section 2.9.2.1, last paragraph before Containment Leakage, the 1st sentence	"As a result of these changes, the licensee calculated larger EAB, LPZ, and CR doses and therefore the changes are shown to be conservative."	The EAB and LPZ doses dropped, only the CR dose increased. Suggested change to "The licensee calculated a larger CR dose due to the changes shown and, although the EAB and LPZ doses decreased slightly due to the new source term, the changes are shown to be conservative." The basis for this proposed change is contained in the license amendment request, attachment 5, Licensing Report, Table 2.9.2-5.
194	Section 2.9.2.1, Containment Leakage, fifth sentence	"The containment leak rate La, and the bypass leak rate are reduced by one-half at 24 hours for offsite calculations, and at 1 hour for CR dose calculations."	Please delete "and the bypass leak rate." Bypass leak rate is included in L_a. The basis for this proposed change is contained in the license amendment request, attachment 5, Licensing Report, Section 2.9.2.1.2.8.
196	Section 2.9.2.1, Releases from the RWST due to ECCS Back Leakage, fourth paragraph, fourth sentence	"The licensee calculated a DF of 450 for the release of iodines from the RWST as a result of back leakage."	Please change "450" to "330". The basis for this proposed change is contained in DNC response to RAI question AADP-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, page 40 of 95.

Page	Location	Sentence	Comment
201	Section 2.9.2.3, third paragraph, third and fourth sentences	"The 35.75 hour time period includes 24 hours for the primary system to cool sufficiently to allow an alignment to the RHRS. As a result of the assumption of a concurrent LOOP, the 35.75 hour period includes an additional 11.75 hours of steaming required to reduce the system heat load to the point where the RHRS can remove all the decay heat using only safety grade equipment."	Please replace with: "The 35.75 hour time period includes 24 hours for the primary system to cool sufficiently to allow an alignment to the RHRS, an additional 11.75 hours of steaming required to reduce the system heat load to the point where the RHRS can remove all the decay heat using only safety grade equipment, and the impact of the assumption of a concurrent LOOP." The basis for the proposed change is as follows: The discussion from DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Page 53 of 95 - 2nd paragraph - The 11.75 hours is not the result of the concurrent LOOP assumption, but the 35.75 hours includes the impact of the concurrent LOOP assumption
203	Section 2.9.2.3, second paragraph, last sentence	"The licensee assumed the same CR ventilation timing sequence as was used for the LOCA, which does not credit automatic initiation of the CREVS."	Please replace with: "The licensee assumed the CR ventilation timing sequence indicated in Table 7, which does not credit automatic initiation of the CREVS." The basis for the proposed change is as follows: The CR ventilation timing sequences for the LOCA and the SGTR are very similar. However, isolation of the Control Room for the SGTR is the result of a radiation monitor (RM) responding to the plume and automatically initiating the CBI signal. The RM and CBI signals each take about 5 seconds. As a

Page	Location	Sentence	Comment
			result of relying on the RM signal there is a 10 second period after the plume reaches the control room intake before the control room is isolated. During the LOCA, the CBI signal is generated and the control room is isolated in 5 seconds which is before the plume reaches the control room intake. Therefore, as a result of the RM signal delay the manual initiation of CREVS occurs 5 seconds later during the SGTR than it does in the LOCA. In Table 3.3-4 of DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Page 62 of 95 - Dominion indicated that the Control Room Ventilation Timing was equal to the current licensing basis FHA due to isolation based on RM not the LOCA and isolation based on SI.
205	Section 2.9.2.4, 1st paragraph, 2nd sentence	"In order to ensure proper accounting of gross gamma, iodine and noble gas releases from the unaffected SGs, the licensee evaluated all the significant nuclide transport models for the MSLB accident. The licensee evaluated the release of the gross gamma activity from the primary coolant, at the TS limit of 100/E-Bar, leaking into the unaffected SG volume at a primary-to-secondary leak rate of 0.65 gpm. Radionuclides initially in the SG liquid and those entering the SG	Dominion did not use a TS limit of 100/Ebar. The gross gamma activity is based on the equivalent level of fuel damage as 1 μCi/gm DE I-131 (0.29% FF). The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, page 54 of 95. Please consider replacing the sentence with: "In order to ensure proper accounting of gross gamma, iodine and noble gas releases from

Page	Location	Sentence	Comment
		from the primary-to-secondary leakage flow are released as a result of secondary liquid steaming. An assumed PC of 100 results in 1 percent of the particulates and iodines in the SG bulk liquid being released to the environment at the steaming rate. Radionuclides initially in the steam space do not provide any significant dose contribution. The transport to the environment of noble gases from the primary coolant is assumed to occur without any mitigation or holdup."	the unaffected SGs, the licensee evaluated all the significant nuclide transport models for the MSLB accident. The licensee evaluated the release of the gross gamma activity from the primary coolant, at a level of fuel failure consistent with 1 uCi/gm DE I-131, leaking into the unaffected SG volume at a primary-to-secondary leak rate of 0.65 gpm. Radionuclides initially in the SG liquid and those entering the SG from the primary-to-secondary leakage flow are released as a result of secondary liquid steaming. An assumed PC of 100 results in 1 percent of the particulates and iodines in the SG bulk liquid being released to the environment at the steaming rate. Radionuclides initially in the steam space do not provide any significant dose contribution. The transport to the environment of noble gases from the primary coolant is assumed to occur without any mitigation or holdup."
206	Section 2.9.2.5, third paragraph, last sentence	"The licensee assumed the same CR ventilation timing sequence as was used for the LOCA, which does not credit automatic initiation of the CREVS."	Please consider replacing with: "The licensee assumed the same CR ventilation timing sequence as was used for the SGTR as shown in Table 7, which does not credit automatic initiation of the CREVS." As with the SGTR, the CR ventilation timing sequence for the LOCA and the LRA are very similar, but the SGTR and LRA CR ventilation timing sequences are identical. Isolation of the

Page	Location	Sentence	Comment
			Control Room for the LRA is the result of a radiation monitor (RM) responding to the plume and automatically initiating the CBI signal. The RM and CBI signals each take about 5 seconds. As a result of relying on the RM signal there is a 10 second period after the plume reaches the control room intake before the control room is isolated. During the LOCA, the CBI signal is generated and the control room is isolated in 5 seconds which is before the plume reaches the control room intake. Therefore, as a result of the RM signal delay the manual initiation of CREVS occurs 5 seconds later during the LRA than it does in the LOCA. The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Table 3.3-4, Page 81 of 95. Just like the SGTR, Dominion indicated that the Control Room Ventilation Timing was equal to the current licensing basis FHA due to isolation based on RM not the LOCA and isolation based on SI.

Page	Location	Sentence	Comment
207	Section 2.9.2.6, second paragraph, first sentence	"The licensee has determined that containment sprays will not initiate due to an REA and, as a result, the licensee did not evaluate dose contributions from ECCS leakage and RWST back leakage as in the LOCA analysis."	Please change "determined' to "conservatively assumed." This assumption did not change from the current licensing bases and is listed as such in the license amendment request, attachment 5, licensing report, page 2.9-16, section 2.9.2.1.6.3 and RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, page 85 of 95, No. 2. Further, it should be noted that ECCS leakage and RWST back leakage are dose contributors when containment recirculation is initiated. Containment recirculation is not necessary for the rod ejection event. This is unrelated to spray operation.
211	Control Room Habitability, fifth paragraph, second sentence	"The period when the CREVS is operational is referred to as the CR positive pressure period."	Please replace with: "The period when the CREVS is providing pressurization flow is referred to as the CR positive pressure period." As stated in the first sentence CREVS is pressurization and recirculation. After 30 minutes following a FHA, only recirculation flow is operating and the control room is neutral until 101 minutes. The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Table 3.2-1, page 48 of 95

Page	Location	Sentence	Comment
215	Table 2, (pg 1 of 2)	Millstone Stack[(1)] 0-2 hrs 2-8 hours.	The Millstone Stack X/Q are listed for 0 – 2 and 2 – 8 hours. They should be 0 – 4 and 4 – 8 hours. The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, page 11 of 95.
223	Table 8, The third item in the table	Primary-to-secondary leak rate TS limit 1 gpm (to unaffected SGs)	Please replace with: Primary-to-secondary leak rate TS limit 150 gpd to any 1 steam generator The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, page 68 of 95.

Page	Location	Sentence	Comment
223	Table 8, last line	"CR plume and CR filter shine dose conservatively set at values from the LOCA analyses"	The MSLB does not use LOCA values, please replace with: "CR plume and CR filter shine dose are 4.37E-03 rem and 0.26 rem, respectively." The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 3, pg 67 of 106.
219	Table 5, (page 2 of 2)	"CR ventilation timing for the LOCA, SGTR, MSLB, LRA and the REA:"	Please delete: "for the LOCA, SGTR, MSLB, LRA and the REA." Each table has its own ventilation timing assumptions.

On Page 219, Table 5, (page2 of 2):

Please replace the following Section:

CR ventilation timing for the LOCA, SGTR, MSLB, LRA and the REA:

T= 0 seconds	Normal CR unfiltered intake flow: 1595 cfm
T= 5 seconds	CBI signal generated
T= 10 seconds	CR isolates on radiation monitor signal Intake flow: 0 cfm; neutral condition Assumed unfiltered inleakage: 350 cfm
T= 1 minute, 5 seconds	delay for CREPS response (Not credited) Assumed unfiltered inleakage: 350 cfm
T=1 hour, 41 min, 5 sec (1.685 hours)	CREVS filtered intake flow: 230 cfm Assumed unfiltered inleakage: 100 cfm CREVS filtered recirculation flow: 666 cfm

With:

CR ventilation timing for the LOCA, SGTR, MSLB, LRA and the REA:

T=0 seconds	CR isolated on SI signal Intake flow: 0 cfm; neutral condition Assumed unfiltered inleakage: 350 cfm
T=1 minute	delay for CREPS response (Not credited) Assumed unfiltered inleakage: 350 cfm
T=1 hour, 41 min (1.683 hours)	CREVS filtered intake flow: 230 cfm Assumed unfiltered inleakage: 100 cfm CREVS filtered recirculation flow: 666 cfm

The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 3, Page 5 of 106; Also stated correctly on page 212, first full paragraph of the draft SER.

On Page 221, Table 7 (Page 1 of 2)

Please replace the following Section:

Table 7 (Page 1 of 2)
MPS3 SPU Data and Assumptions for the SGTR Accident

Primary-to-secondary leak rate TS limit		1 gpm (to unaffected SGs)
LOOP		Coincident with release
RCS TS iodine limit for normal operation		
	Gross gamma	100 / E-Bar
	Iodine	1.0 μCi/gm DEI

With:

Table 7 (Page 1 of 2)
MPS3 SPU Data and Assumptions for the SGTR Accident

Primary-to-secondary leak rate TS limit	1 gpm (to unaffected SGs)
LOOP	Coincident with release
RCS TS iodine limit for normal operation	1.0 μCi/gm DEI
RCS Gross Gamma activity	Equivalent to fuel failure associated with DEI limit

Basis:
The SGTR does not use the 100/E-bar LCO. Instead, as indicated in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Page 54 of 95, the gross gamma activity was based

upon the more operationally limiting Dose Equivalent I-131 LCO. The discussion from Letter Serial No. 07-0846 Attachment 2, Page 54 of 95 is repeated below.
"Primary side iodine and gross gamma source concentrations are based on the more limiting Technical Specification limit of 1.0 µCi/gm Dose Equivalent (DEQ) I-131. The failed fuel equivalent associated with the iodine Technical Specification limit is 0.29%. The Technical Specification concentrations of the non-iodine isotopes at the 100/E-bar limit result in greater than 1 % failed fuel. Therefore the more limiting failed fuel percentage associated with the specific activity limit for the iodines is used for all primary coolant isotopes."

On Page 223, Table 8, 4th item on the table:
Please replace the following Section:

Table 8 - MPS3 SPU Data and Assumptions for the MSLB Accident

RCS volume	11,750 ft3
RCS mass	5.216E+05 lbm
Primary-to-secondary leak rate TS limit	1 gpm (to unaffected SGs)
RCS TS limit for normal operation	
Gross gamma	100/ E-Bar
Iodine	1.0 µCi/gm DEI

With:

Table 8 - MPS3 SPU Data and Assumptions for the MSLB Accident

RCS volume	11,750 ft3
RCS mass	5.216E+05 lbm
Primary-to-secondary leak rate TS limit	1 gpm (to unaffected SGs)
RCS TS iodine limit for normal operation	1.0 µCi/gm DEI
RCS Gross Gamma activity	Equivalent to fuel failure associated with DEI limit

Basis:

Dominion did not use a technical specifications limit of 100/Ebar. The gross gamma activity is based on the equivalent level of fuel damage as 1 μCi/gm DE I-131 (0.29% FF). The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Page 54 of 95 (See comment 28)

Page	Location	Sentence	Comment
222	Table 7 (Page 2 of 2)	“CR ventilation timing for the SGTR: Same as the LOCA”	Please delete "Same as the LOCA". The CR ventilation timing sequences for the LOCA and the SGTR are very similar. However, isolation of the Control Room for the SGTR is the result of a radiation monitor (RM) responding to the plume and automatically initiating the CBI signal. The RM and CBI signals each take about 5 seconds. As a result of relying on the RM signal there is a 10 second period after the plume reaches the control room intake before the control room is isolated. During the LOCA, the CBI signal is generated and the control room is isolated in 5 seconds which is before the plume reaches the control room intake. Therefore, as a result of the RM signal delay the manual initiation of CREVS occurs 5 seconds later during the SGTR than it does in the LOCA. The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-0846), Attachment 2, Page 62 of 95. DNC indicated that the Control Room Ventilation Timing was equal to the current licensing basis FHA due to isolation based on RM not the LOCA and isolation based on SI.
223	Table 8 (Page 1 of 1)	“CR ventilation timing for the MSLB: Same as the LOCA”	Please delete "Same as the LOCA". The basis for the proposed change is contained in RAI Response to question AADB-07-0107, DNC letter dated January 18, 2008 (Serial No. 07-

			0846), Attachment 2, Page 72 of 95. DNC indicated that the Control Room Ventilation Timing was equal to the current licensing basis FHA.
224	Table 9 (Page 1 of 1)	"CR ventilation timing Same as the LOCA"	Please replace with: "CR ventilation timing Same as the SGTR". The CR ventilation timing sequences for the LOCA and the SGTR are very similar. However, isolation of the Control Room for the SGTR is the result of a radiation monitor (RM) responding to the plume and automatically initiating the CBI signal. The RM and CBI signals each take about 5 seconds. As a result of relying on the RM signal there is a 10 second period after the plume reaches the control room intake before the control room is isolated. During the LOCA, the CBI signal is generated and the control room is isolated in 5 seconds which is before the plume reaches the control room intake. Therefore, as a result of the RM signal delay the manual initiation of CREVS occurs 5 seconds later during the SGTR than it does in the LOCA.

Section 2.10, Human Performance

Page	Location	Sentence	Comment
227	Changes in Emergency and Abnormal Operating Procedures, second bullet, First sentence.	"There will be an additional manual action in the EOPs to verify ECCS flow when RCS pressure is less than 1900 psig."	Please replace "psig" with "psia". The unit "psig" was incorrectly used on page 2.11-3 of the SPU licensing report, section 2.11.1, Human Factors. However, the correct units were used in reference to same parameter on pages 2.4-29, 2.4-30, 2.4-39 and 2.4-40 of the SPU licensing report.

Section 3.2, Technical Specifications

Page	Location	Sentence	Comment
234	Last sentence	"This surveillance needs to be performed within 24 hours after reaching 90 percent of the RTP."	Please replace with: "This surveillance needs to be performed within 24 hours after reaching 90 percent of the RTP following each fuel loading." This change will make the sentence consistent with the revised surveillance requirement.
235	First full sentence	"Thus, to comply with the uncertainty analysis assumptions and to meet the heat balance requirements, it is necessary to require the RCS flow rate to be determined at not less than 90 percent of the RTP and within 24 hours after reaching 90 percent of the RTP."	Please replace with: "Thus, to comply with the uncertainty analysis assumptions and to meet the heat balance requirements, it is necessary to require the RCS flow rate to be determined at not less than 90 percent of the RTP and within 24 hours after reaching 90 percent of the RTP following each fuel loading." This change will make the sentence consistent with the revised surveillance requirement.
235	Second paragraph, third sentence	"As described above, the instrumentation calibrations required for heat balance are calibrated once per 18 months and within 24 hours after reaching 90 percent of the RTP."	Please replace with: "As described above, the instrumentation calibrations required for heat balance are calibrated once per 18 months and within 7 days prior to performance of the calorimetric flow measurement." This sentence is describing the current surveillance requirement.
236	TS 3/4.3.2 Engineered Safety Features Actuation System Instrumentation, second paragraph, third line.	"Pressurizer Water Level-Low"	Please replace the word "Water Level" with "Pressure" to read: "Pressurizer Pressure-Low". Please refer to Technical Specifications Table 2.2-1, Functional Unit 9.

Page	Location	Sentence	Comment
239	Additional Changes, Item 1, Safety Grade Cold Shutdown, second paragraph.	"The licensee's application dated July 13, 2007, provided revised TS Bases pages to be implemented with the associated TS changes. The NRC staff reviewed this change in Section 2.5 of the SE and did not have any concerns. The NRC staff notes that the MPS3 TS Bases Control Program is the appropriate process for updating the TS Bases."	Please delete this paragraph. The licensing bases change associated with the proposed change is contained in FSAR Section 5.4.7.2.3.5 as addressed in MPS3 SPU license amendment request, attachment 1, section 3.1, page 10, last paragraph.
240	Additional Changes, Item 2, BTP CMEB 9.5.1 Sections 5.c.3 and 5.c.5-Fire shutdown strategy for long-term steam generator inventory make-up, second paragraph.	"The licensee's application dated July 13, 2007, provided revised TS Bases pages to be implemented with the associated TS changes. The NRC staff reviewed this change in Section 2.5 of the SE and did not have any concerns. The NRC staff notes that the MPS3 TS Bases Control Program is the appropriate process for updating the TS Bases."	Please delete this paragraph. The licensing bases change associated with the proposed change is contained in a DNC document entitled: "MP3 Branch Technical Position 9.5-1 Compliance Report." This document will be updated upon NRC approval of this item to credit the use of domestic water, demineralized water or fire water to make-up the DWST and CST.