Dominion Nuclear Connecticut, Inc.
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Dominion®

January 11, 2008

Withhold Attachment 1 from Public Disclosure Under 2.390(a)(4)

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

Serial No.: 07-0834H
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 QUESTION EMCB-07-0070

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 question EMCB-07-0070 of this RAI is provided in Attachment 1 to this letter.

Attachment 7 to the LAR submitted in the July 13, 2007 letter (Serial No. 07-0450), contained information proprietary to Westinghouse Electric Company LLC (Westinghouse). Attachment 6 to that July 13, 2007 letter contained an affidavit dated June 25, 2007, signed by Westinghouse, the owner of the information, requesting that the information in Attachment 7 be withheld from public disclosure. The NRC determined that this information should be withheld from public disclosure as noted by the NRC letter from Mr. John G. Lamb to Mr. James A. Gresham, Westinghouse, dated October 18, 2007. Similarly, the response to question EMCB-07-0070 has been determined to contain proprietary information by Westinghouse, the owner of the information. For the reasons previously set forth in the affidavit by J. A. Gresham, Manager Regulatory Compliance and Plant Licensing, Westinghouse requests that the NRC treat the information in Attachment 1 to this letter as proprietary to Westinghouse and accordingly, that this information be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4).

To conform to the requirements of 10 CFR 2.390 concerning the protection of proprietary information, the proprietary information provided in Attachment 1 is contained within brackets. Attachment 2 has been redacted to provide a non-proprietary version of the requested information. Where the proprietary information has been deleted in the non-proprietary version only the brackets remain (i.e., the information that was contained within the brackets in the proprietary version has been redacted).

A001
NRR

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference Westinghouse letter CAW-07-2296 and should be addressed to Mr. J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

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
140842
My Commission Expires May 31, 2010

Commitments made in this letter: None

Attachments

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

Mr. J. G. Lamb
U.S. Nuclear Regulatory Commission
One White Flint North
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Ms. C. J. Sanders
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Mr. S. W. Shaffer
NRC Senior Resident Inspector
Millstone Power Station

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

ATTACHMENT 2

LICENSE AMENDMENT REQUEST

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RESPONSE TO QUESTION EMCB-07-0070

REDACTED VERSION

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

Mechanical and Civil Engineering Branch

EMCB-07-0070

Section 2.2.3.2.1 states that "Changes in the primary coolant system operating conditions (e.g., increase in power) also produce changes in the boundary conditions; this includes loads and temperatures experienced by the reactor internals structures or components. Ultimately, this results in changes in the stress levels in these components and changes in the relative displacement between the reactor vessel and the reactor internals. To ensure that the reactor internal components maintain their design functions, and to ensure safety questions have been reviewed, a systematic evaluation of the reactor components has been performed to assess the impact of increased core power on the reactor internal components." Table 2.2.3-3 contains a summary of stresses and fatigue usage factors for core support structures. Confirm that these values are for SPU conditions and provide corresponding values at current conditions.

DNC Response

The values presented in Table 2.2.3-3 of the MPS3 Licensing Report are for the SPU conditions. The corresponding values at current conditions have been added to the table shown below.

Table 2.2.3-3
Reactor Internal Components Stresses and Fatigue Usage Factors

Component	Current Stress Intensity (ksi) S.I. = (P_m + P_b + Q)	SPU Stress Intensity (ksi) S.I. = (Pm + Pb + Q)	Allowable S.I. (3 S_m) ksi	Fatigue Usage
Upper Core Plate	[ ]$^{(3)\ a,c}$	[ ]^{a,c}	48.6	[ ]^{a,c}
Lower Support Plate	[ ]^{a,c}	[ ]^{a,c}	48.3	[ ]^{a,c}
Lower Core Plate	[ ]$^{(3)\ a,c}$	[ ]^{a,c}	48.6	[ ]^{a,c}
Lower Support Columns	[ ]^{a,c}	[ ]^{a,c}	48.3	[ ]^{a,c}
Core Barrel Outlet Nozzle: Section A-A Section B-B	 [ ]$^{(1)\ a,c}$ [ ]^{a,c}	 [ ]$^{(1,4)\ a,c}$ [ ]$^{(5)\ a,c}$	 34.4 49.2	 [ ]$^{(4)\ a,c}$ [ ]^{a,c}
Baffle-Former Bolts$^{(2)}$	--	--	--	--

Notes:
1) Exceeded 3 S_m limit, simplified elastic-plastic analysis was performed to calculate fatigue strength, as allowed by ASME, B&PV Code, Section III, NG 3228.3. These conditions have been met and the fatigue usage is less than 1.0.
2) The basis of the baffle-former bolt qualification is a fatigue test. The evaluation of the revised loads consisted of demonstrating that the loads associated with SPU are acceptable for the plant design life. Therefore, it is concluded that the baffle-former bolts are structurally adequate for the SPU RCS conditions.
3) The current upper and lower core plate stress intensities are based on two-dimensional analysis. The upper and lower core plate analysis for the Millstone 3 SPU project are based on three-dimensional finite element models with update heating rates that supersede those used in the current upper and lower core plate analysis.
4) SPU stress intensity changed from [ ]^{a,c} ksi to [ ]^{a,c} ksi and fatigue usage changed from [ ]^{a,c} to [ ]^{a,c} due to a change in method used to combine the stresses.
5) SPU stress intensity changed from [ ]^{a,c} ksi to [ ]^{a,c} ksi due to a change in method used to combine the stresses.

The MPS3 reactor internal components were designed and built prior to the implementation of Subsection NG of the ASME Code Section III; therefore, no plant-specific ASME Code stress report was written for the reactor internal components. The MPS3 reactor internal components were analyzed to meet the intent of the ASME Code,

Code, Section III 1971 Edition with Addenda through Summer 1973 criteria. But based on the previous evaluations and current practices, the guidance in Subsection NG of the ASME Code was used for this evaluation.