Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUN 3 2003

Docket No. 50-336
B18917

RE: 10 CFR 50, Appendix H, III.B.3

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Power Station, Unit No. 2
Changes to the Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule

Dominion Nuclear Connecticut, Inc. (DNC) hereby requests the Nuclear Regulatory Commission (NRC) verification of a proposed revision to the Millstone Unit No. 2 (MP2) Reactor Pressure Vessel (RPV) Surveillance Capsule Withdrawal Schedule that is in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," paragraph III.B.3. This change to the withdrawal schedule conforms to the ASTM E185-82 standard as specified in 10 CFR 50, Appendix H, and is described in detail in Attachment 1. Accordingly, NRC verification of this conformance is requested by October 15, 2003, to support plans for a Fall 2003 refueling outage.

There are no regulatory commitments contained within this letter.

If you should have any questions regarding this submittal, please contact Mr. David W. Dodson at (860) 447-1791, ext 2346.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Attachment (1)

cc: H. J. Miller, Region I Administrator
R. B. Ennis, NRC Senior Project Manager, Millstone Unit No. 2
NRC Senior Resident Inspector, Millstone Unit No. 2

A008

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Attachment 1

Millstone Power Station, Unit No. 2
Changes to the Reactor Pressure Vessel
Surveillance Capsule Withdrawal Schedule

Millstone Power Station, Unit No. 2
<u>Changes to the Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule</u>

BACKGROUND:

NRC Administrative Letter (AL) 97-04, "NRC Staff Approval for changes to 10 CFR 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules," was issued to inform licensees that changes to facilities' reactor vessel surveillance specimen capsule withdrawal schedules that do not conform to the required ASTM standard referenced in 10 CFR 50 Appendix H be treated as license amendments. However, changes to withdrawal schedules that meet the applicable ASTM standard do not exceed the operating authority already granted in the plant's license and therefore, no license amendment is required per AL 97-04. The change still requires the NRC verification of conformance with the ASTM standard.

In this proposal, no license amendment is required because the proposed revision to the Millstone Unit No. 2 (MP2) Reactor Pressure Vessel (RPV) Capsule Withdrawal Schedule, described in the balance of this attachment, meets the applicable ASTM standard requirements. Pursuant to 10 CFR 50, Appendix H, paragraph III.B.3, any changes must be submitted for review and approval. Accordingly, DNC requests that the NRC provide verification of this conformance.

CURRENT SCHEDULE:

The current reactor vessel material surveillance withdrawal schedule is shown in Tables 1 and 2, and is contained in both the Technical Requirements Manual (TRM) and the Updated Final Safety Analysis Report (FSAR).

Table 1
Current Reactor Vessel Surveillance Capsule Withdrawal Schedule
(as provided in the FSAR TABLE 4.6-9)

Location on Vessel Wall, Degrees	Removal Schedule, Effective Fully Power Years (EFPY)	Fluence, n/cm²
97	(Pulled in 1982) 3	3.78×10^{18} *
104	(Pulled in 1990) 10	8.84×10^{18} *
284	17	3.0×10^{19}
263	24	4.8×10^{19}
277	32	6.6×10^{19}
83	Spare	
97° Flux Monitor (installed at BOC 6)	(Pulled in 1990) 10	7.04×10^{18} *

* Actual data

Table 2
Current Reactor Vessel Surveillance Capsule Withdrawal Schedule
(as provided in the TRM Table 4.4-3)

Capsule	Schedule (EFPY)
W-97	3.0
W-104	10.0
W-284	17.0
W-263	24.0
W-277	32.0
W-83	Spare
W-97 (Flux Monitor)	10.0

PROPOSED SCHEDULE:

Table 3
Proposed Reactor Vessel Surveillance Capsule Withdrawal Schedule

Capsule	Location	Lead Factor [(a)]	Removal Time (EFPY) [(b)]	FLUENCE (n/cm^2, E>1.0MeV) [(a)]
W-97	97°	1.40	3.0	3.24×10^{18} [(c)]
W-104	104°	0.95	10.0	9.49×10^{18} [(c)]
W-97 [(d)]	97°		10.0	- -
W-83	83°	1.31	15.3	1.74×10^{19} [(c)]
W-277	277°	1.31	EOL[(e)]	[(e)]
W-263	263°	1.31	Standby	- -
W-284	284°	0.97	Standby	- -

Notes:
(a) Updated in Capsule W-83 dosimetry analysis.
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Plant specific evaluation.
(d) Flux Monitor
(e) EOL is defined as the end-of-license period corresponding to the original 40 year license. Capsule W-277 is projected to receive 1.31 times the reactor vessel peak EOL surface fluence of 2.40×10^{19} n/cm^2 (E>1.0MeV). Capsule W-277 will receive the vessel peak EOL surface fluence at 23.2 EFPY. It will be removed before it receives twice the peak vessel surface fluence of 4.80×10^{19} n/cm^2 (E>1.0MeV).

JUSTIFICATION:

The proposed schedule reflects the incorporation of the most recent surveillance capsule analysis provided by DNC's letter dated February 26, 2003.[1] Capsule W-83, identified as a standby capsule, was removed based upon better agreement with ASTM E185-82 and utilization of a capsule with a higher lead factor.

Development of the new withdrawal schedule is based upon ASTM E185-82.[2] Use of ASTM E185-82 was selected to meet the requirements of 10 CFR 50, Appendix H. Consideration is given to the predicted transition temperature shift and the reactor vessel inside surface. The maximum transition temperature of limiting material is between 100°F and 200°F. (Note: No beltline material is predicted to fall below 50 ft-lb at the 1/4 thickness (1/4t) location). Based on ASTM E185-82 requirements, removal of four (4) capsules is recommended.

Historically, the first capsule, W-97, was removed and evaluated after 3.0 EFPY. The results were documented in TR-N-MCM-008.[3] The second capsule W-104 and supplemental dosimetry (W-97 Flux Monitor) were removed after 10 EFPY. The results are documented in BAW-2142.[4] This information is currently contained in Table 4.6-9 of the FSAR and in Table 4.4-3 of the TRM. The most recent capsule evaluated (W-83) was the third capsule removed. Its removal was after 15.3 EFPY and is documented in WCAP-16012,[5] consistent with the ASTM Standard recommendation of 15 EFPY.

In selection of the fourth capsule, it is beneficial to utilize the surveillance capsules with the highest lead factors earlier in life. Therefore, capsules W-263 or W-277 were considered given they both have projected lead factors of 1.31. Based upon the lead factor, the capsule will receive between one and two times the end-of-life vessel surface fluence at end-of-life, which is consistent with ASTM E185-82 recommendations. Consideration was then given to the capsule contents. Specifically, while both contain relevant materials, W-263 contains SRM or "standard reference material" while W-277 contains transverse charpy impact base material specimens from the limiting MP2 material. This standard reference material is not a limiting material associated with

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(1) Millstone Nuclear Power Station, Unit No. 2, "Submittal of Third Reactor Vessel Surveillance Capsule Report," February 26, 2003, (Accession No. ML030660131).

(2) ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)," 1982.

(3) "Evaluation of Irradiated Capsule W-97," Report No. TR-N-MCM-008, Combustion Engineering, Inc., April 1982.

(4) "Analysis of Capsule W-104 Northeast Nuclear Energy Company Millstone Nuclear Power Station, Unit No. 2," Report No. BAW-2142, B&W Nuclear Service Company, November 1991.

(5) "Analysis of Capsule W-83 from the Dominion Nuclear Connecticut Millstone Unit No. 2 Reactor Vessel Radiation Surveillance Program," Report No. WCAP-16012, Westinghouse, February 2003, (Accession Nos. ML030710172 and ML030660170).

MP2. It provides additional information relative to neutron irradiation damage but it is also less useful in terms of specific damage for MP2 materials. Therefore, capsule W-277 was selected as the fourth capsule for removal. Capsules W-263 and W-284 will be identified as spare and available for supplemental testing or license renewal activities.

PRECEDENCE:

Precedence for this type of request and NRC confirmation of a revised capsule withdrawal schedule is in a September 13, 2001,[6] letter submitted by FirstEnergy Nuclear Operating Company, and NRC approval letter dated March 19, 2002.[7] Precedence is also in letters dated April 26,[8] and July 14, 2000,[9] for a PECO Energy Company request.

CONCLUSION:

The proposed revision to the RPV capsule withdrawal schedule meets the applicable ASTM standard requirements that are referenced in Appendix H to 10 CFR 50 and described above. Accordingly, NRC verification of this conformance is requested by October 15, 2003, to support plans for a Fall 2003 refueling outage.

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(6) FirstEnergy Nuclear Operating Company letter, "Beaver Valley Power Station, Unit No. 2, Docket No. 50-412, License No. NPF-73, Reactor Vessel Capsule W Test Report," September 13, 2001. (Accession No. ML012700020)

(7) NRC letter, "Beaver Valley Power Station, Unit 2 – Changes to the Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule (TAC No. MB2974)," March 19, 2002. (Accession No. ML020780425)

(8) PECO Energy Company letter, "Peach Bottom Atomic Power Station, Units 2 and 3, Revision to the Specimen Capsule Withdrawal Schedule," April 26, 2000. (Accession No. ML003712027)

(9) NRC letter, "Request by PECO Energy Company to Modify the Peach Bottom Atomic Power Station, Units 2 and 3, Reactor Vessel Surveillance Capsule Withdrawal Schedules (TAC Nos. MA8901 and MA8902)," July 14, 2000. (Accession No. ML003727917)