March 29, 2007

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3, ISSUANCE OF AMENDMENT RE: DELETING REDUNDANT SURVEILLANCE REQUIREMENTS PERTAINING TO POST MAINTENANCE/POST MODIFICATION TESTING (TAC NO. MD0693)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 237 to Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3. This amendment consist of changes to Facility Operating License No. NPF-49 in response to your application dated March 28, 2006, as supplemented by letters dated October 26, and December 4, 2006, and January 26, 2007.

This amendment revises the Technical Specifications to delete redundant surveillance requirements pertaining to post-maintenance/post-modification testing.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/ra/

Victor Nerses, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 237 to License No. NP-49
2. Safety Evaluation

cc w/encls: See next page

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OFFICE	LPLI-2/PM	LPLI-2/LA	SCVB/BC	SPWB/BC	ITSB/BC	EEEB/BC	OGC	LPLI-2/BC
NAME	VNerses GMiller for	CSola	RDennig*	JNakoski*	TKobetz	GWilson*	DRoth	HChernoff
DATE	3/29/07	3/15/07	02/15/07	12/05/07	3/15/07	2/22/07	3/29/07	3/29/07

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DOMINION NUCLEAR CONNECTICUT, INC

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Dominion Nuclear Connecticut, Inc. (the licensee) dated March 28, 2006, as supplemented by letters dated October 26, and December 4, 2006, and January 26, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 237, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. Dominion Nuclear Connecticut, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: March 29, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 237

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace page 4 of Facility Operating License No. NPF-49 with the attached revised page 4. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3/4 1-25	3/4 1-25
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6
3/4 6-19	3/4 6-19
3/4 7-16	3/4 7-16
3/4 7-20	3/4 7-20
3/4 8-7	3/4 8-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING

LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated March 28, 2006, as supplemented by letters dated October 26, and December 4, 2006, and January 26, 2007, Dominion Nuclear Connecticut, Inc. (DNC, or the licensee), requested a license amendment for Millstone Power Station, Unit No. 3 (MPS3). The licensee requested to delete redundant Technical Specifications (TS) surveillance requirements pertaining to post-maintenance/post-modification testing

The supplements dated, October 26, and December 4, 2006, and January 26 , 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards determination as published in the *Federal Register* on May 23, 2006 (71 FR 29673).

2.0 REGULATORY EVALUATION

The following Nuclear Regulatory Commission (NRC, or the Commission) regulatory requirements and Regulatory Guides (RGs) are applicable to this TS change application:

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36 sets forth the regulatory requirements relating to the content of TSs. Specifically, 10 CFR 50.36(c)(3) lists "Surveillance Requirements" as one of the required items that must be included in the TS, and states that "surveillance requirements are requirements relating to test, calibration, or inspection to insure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation (LCO) will be met."

General Design Criterion (GDC) 17, "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR, Part 50, states, in part, that an onsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The onsite electric power supplies and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of the loss of power from the unit, the transmission network, or the onsite electric power supplies.

GDC 18, "Inspection and Testing of Electric Power Systems," states that electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing to demonstrate operability and functional performance.

Since NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," was developed based on the 10 CFR 50.36 requirements for Westinghouse plants, and MPS3 is a plant designed by Westinghouse company, the NRC staff utilizes the NUREG-1431 guidance in its review of the proposed TS changes for MPS3. The specific sections of the standard TSs (STS) used are STS SR 3.0.1, "SR Applicability," and its associated Bases section, STS 3.1.4, "Rod Group Alignment Limits," and STS 3.5.2, "ECCS - Operating."

RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", Revision 2, March 1978.

RG 1.108, Revision 1, August 1977, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," (superceded by RG 1.9, Revision 3 in 1993), describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to periodic testing of diesel electric power units to ensure that the diesel electric power systems will meet their availability requirements.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the license amendment request (LAR) to delete or modify the following post-maintenance or post-modification SRs:

3.1 Deletion of SR 4.1.3.4.b

The existing SR 4.1.3.4 requires verification of control rod drop times to provide assurance that the maximum allowable control rod drop time is bounded by the assumed control rod drop time used in the safety analysis. Measuring control rod drop times prior to reactor criticality, after reactor vessel head removal and installation, ensures that the reactor internals and control rod drive mechanism (CRDM) will not interfere with rod motion or control rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. Specifically, Item b of SR 4.1.3.4 requires verification of the allowable maximum control rod time of certain control rods through measurement prior to reactor criticality, following any maintenance on, or modification to, the CRDM system which could affect the drop time of those specific rods. In the LAR, the license proposed to delete SR 4.1.3.4.b that states that "b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of these specific rods, and..."

3.2 Revision to SR 4.5.2.g.1

The ECCS is designed to cool the reactor core and provide shutdown capability during various accident conditions. The MPS3 ECCS components are designed such that a minimum of three accumulators, one charging (CH) pump, one safety injection (SI), one residual heat removal

(RHR) pump, one containment recirculation pump, and one containment recirculation cooler and their associated valves and piping are available to provide adequate core cooling during a design-basis loss-of-coolant accident (LOCA).

The existing SR 4.5.2 is provided to ensure operability of each ECCS component such that at a minimum, the assumptions used in the safety analyses are met. SR 4.5.2.g.1 for throttle valve position stops provides assurance that proper ECCS flow will be maintained in the event of the LOCA. The proposed TS revises SR 4.5.2.g.1 by deleting the wording: "or maintenance on the valves" from SR 4.5.2.1.g that states that "within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE..."

3.3 Deletion of SR 4.5.2.h

The existing SR4.5.2.h requires maintenance of proper flow resistence and pressure drop in the piping system (including CH pump, SI pump and RHR pump lines) to each injection point to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration; (2) provide the proper flow split between injection points in accordance with the assumptions used in the LOCA analyses; and (3) provide an acceptable level of total ECCS flow to all injection points equal to or greater than that assumed in the LOCA analyses. In the LAR the licensee proposed to delete SR 4.5.2.h.

In support of the TS changes, the licensee also added the clarifications for post-maintenance testing to the bases of the affected TSs as follows:

- Add to TS Bases 3/4.1 the following clarification -

  Measuring rod times prior to reactor criticality, after reactor vessel head removal and installation, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. Any time the OPERABILITY of the control rods has been affected by a repair, maintenance, modification, or replacement activity, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

- Add to TS Bases 3/4.5 the following clarification -

  Any time the OPERABILITY of the ECCS throttle valves or an ECCS subsystem has been affected by repair, maintenance, modification, or replacement activity that alter flow characteristics, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

The existing SR 4.0.1 specifies the general requirements of the SRs. Its bases clarify the requirement for post-maintenance testing and state, in part, that:

> Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with Specification 4.0.2. Post maintenance testing may not be possible in the current Mode or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

In comparison with NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," the NRC staff found that the existing SR 4.0.1 and its bases are consistent with STS SR 3.0.1 and its bases regarding the required post-maintenance testing. Also, the NRC staff found that the proposed deletion of SRs 4.1.3.4.b, part of 4.5.2.g.1 and 4.5.2.h respectively, is consistent with STS 3.1.4, "Rod Group Alignment Limits," and STS 3.5.2, "ECCS - Operating," of NUREG-1431, which does not contain the SRs proposed to be deleted from the MPS3 TSs..

However, the NRC staff found that the existing SR 4.0.1 does not specify the post-maintenance tests that replace the tests formally specified in the deleted SRs. For the post-maintenance testing requirements, SR 4.1.3.4.b requires the allowable maximum control rod drop time to be demonstrated, SR 4.5.2.g.1 requires verification of the correct position of each electrical and/or mechanical stop or ECCS throttle valves, and SR 4.5.2.h requires flow balance testing to be performed on the charging pump lines, safety injection lines and residual heat removal pump lines.

During the course of the review, the NRC staff requested the licensee to identify relevant maintenance procedures or documents that discuss the required post-maintenance testing, and identify the testing requirements equivalent to the deleted SRs will be performed in accordance with SR 4.0.1. In response, the licensee indicated (Reference (Ref.) 2) that SR 4.1.3.4.b and portions of 4.5.2.g.1 proposed for deletion are included in existing system test procedures. Specifically, system procedure (SP)-3451N22 specifies that the measurement of the control rod insertion time is conducted to meet a specified criterion for specifically affected individual rods following any maintenance on or modification to the control rod drive system that could affect the insertion time of those rods. SP-3713Z specifies that setting position of high pressure safety injection (HPSI) throttle valves is performed within 4 hours after valve maintenance. In the supplemental RAI response (Ref. 3), the licensee confirmed that SP-3451N22 specifies the measured control rod insertion time to be equal to or less than 2.19 seconds. Since this value is the same value used in the safety analysis included in the final safety analysis report the NRC staff determined that the value is acceptable. The licensee also confirmed (Ref. 3) that the HPSI throttle valves specified in SP-3713Z are the same ECCS throttle valves specified in SR 4.5.2.g.2. With respect to SR 4.5.2.h, the licensee agreed (Ref. 2) to preserve the fundamental requirements to perform ECCS flow balance testing to the specified acceptance criteria and relocate the flow requirement for the CH pump lines, SI pump lines and RHR pumps lines (SRs 4.5.2.h.1, 4.5.2.h.2 and 4.5.2.h.3) to the MPS3 technical requirements manual (TRM). The TRM will require the flow balance test to be performed following completion of the modification to the ECCS subsystems that adversely impact the flow rates described in TRM.

Based on its review of the licensee's RAI responses discussed above, the NRC staff found that the frequency and acceptance criteria of the required tests specified in those SPs and TRM are consistent with that of the post-maintenance tests specified in the deleted SRs, therefore, concluded that the licensee's SPs and TRM provide reasonable assurance that the intent of the deleted SRs will be met.

As discussed in the above review, the NRC staff found that the proposed deletion of the SRs is consistent with the guidance in the STS documented in NUREG-1431 for Westinghouse plants, and the required post-maintenance tests specified in the deleted SRs will be adequately specified in the MPS3 system test procedures. Therefore, the NRC staff considers that the proposed deletion of SR 4.1.3.4.b, a portion of SR 4.5.2.g.1, and SR 4.5.2.h is acceptable.

3.4 <u>Revisions to SR 4.6.6.1.b, 4.7.7.c and 4.7.9.b</u>

Since all three proposed SR amendments parallel each other in that they delete from the respective SR the requirements to perform the SR's following any structural maintenance on the HEPA filter or charcoal adsorber housings, all three SR amendments will be analyzed collectively.

3.4.1 For the Supplementary Leak Collection and Release Systems, Surveillance Requirement 4.6.6.1.b currently reads:

> (Note that the underline added to the following SRs denotes the portion that is to be deleted)
>
> SR4.6.6.1 Each Supplementary Leak Collection and Release System shall be demonstrated OPERABLE:
>
> b. At least once per 24 months <u>or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings,</u> or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:
>
> 1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% [percent] and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 7600 cfm to 9800 cfm;

The proposed SR 4.6.6.1.b reads:

> SR4.6.6.1 Each Supplementary Leak Collection and Release System shall be demonstrated OPERABLE:
>
> b. At least once per 24 months or following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 7600 cfm to 9800 cfm;

3.4.2 For the Control Room Emergency Air Filtration Systems, SR 4.7.7.c currently reads:

4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

c. At least once per 24 months <u>or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings,</u> or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 1120 cfm ± 20%;

The proposed SR.7.7.c reads:

SR 4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

c. At least once per 24 months or following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 1120 cfm ± 20%;

3.4.3 For the Auxiliary Building Filter Systems SR 4.7.9.b currently reads:

SR 4.7.9 Each Auxiliary Building Filter System shall be demonstrated OPERABLE:

b. At least once per 24 months <u>or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings,</u> or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions

C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 30,000 cfm ± 10%;

The proposed SR 4.7.9.b reads:

SR 4.7.9 Each Auxiliary Building Filter System shall be demonstrated OPERABLE:

b. At least once per 24 months or following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 30,000 cfm ± 10%;

The proposed changes remove the requirement to verify that the filter system satisfies the acceptance criteria for:

1) the in-place penetration and bypass leakage testing,
2) the carbon sample laboratory analysis and
3) the system flow rate

after any structural maintenance on the HEPA filter or charcoal adsorber housings from SR 4.6.6.1.b, SR 4.7.7.c and SR 4.7.9.b.

The licensee stated in its application of March 31, 2006 as the reason for the removal of this clause from SR 4.6.6.1.b, SR 4.7.7.c and SR 4.7.9.b the following:

SR 4.1.3.4.b, SR 4.5.2.h and the deleted parts of SR 4.5.2.g.1, SR 4.6.6.1.b, SR 4.7.7.c, SR 4.7.9.b and SR 4.8.1.1.2.h describe testing activities which are performed following repair, maintenance, modification, or replacement of a component. These SRs represent a duplication of SR 4.0.1 testing requirements. Any time the operability of a system or component has been affected by repair, maintenance, modification or replacement, post maintenance testing is required to demonstrate the operability of the system or component in accordance with SR 4.0.1. The testing requirements specified in SR 4.0.1 provide an equivalent level of assurance that the affected subsystems will continue to be tested in a manner necessary to give confidence that the subsystems can perform their intended safety function. Therefore, these SRs can be deleted. (Attachment 1, Section 3.6)

On page 3 of the Attachment from the supplement dated January 26, 2007, (Ref. 4) the licensee acknowledges the requirement to demonstrate a subject filter train's operability anytime maintenance activities are performed that could impact the filter train's operability, with the words:

DNC understands the requirement of TS 4.0.1 to apply any time a maintenance activity is conducted and the activity has the potential to impact a TS required

> system, structure or components ability to meet the performance specified in any SR. As such, any maintenance activity affecting the ability of the filter train to meet the bypass acceptance criteria specified in the modified SR would make the previous test results invalid and thereby require the associated SR to be re-performed to verify operability."

The staff concludes that SR 4.0.1 would require the licensee to repeat periodic surveillances on filters and filter housings, including the in place bypass leakage test, as a post maintenance test if the maintenance activity, such as structural maintenance on the housing, could affect the results of the previous surveillance test which determined operability.

Based on this conclusion, the staff finds the proposed changes to SR 4.6.6.1.b, SR 4.7.7.c and SR 4.7.9.b to be consistent with RG 1.52 and is therefore acceptable.

3.5 Revision to SR 4.8.1.1.2.h

SR 4.8.1.1.2.h currently requires that each EDG shall be demonstrated operable at least once per 10 years or after any modifications which could affect EDG interdependence by starting both EDGs simultaneously from standby conditions, during shutdown, and verifying that both EDGs achieve generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 hertz in less than or equal to 11 seconds. SR 4.8.1.1.2.h demonstrates that the EDG starting independence has not been compromised and that each engine can achieve proper speed within the specified time when the EDGs are started simultaneously. SR 4.8.1.1.2.h is in conformance with RG 1.108, which the licensee has committed to in the MPS3 Updated Final Safety Analysis Report.

The licensee has proposed to delete a redundant post-modification surveillance requirement from SR 4.8.1.1.2.h associated with EDG testing. Specifically, the licensee proposed to modify the TS by deleting the phrase, "or after any modifications which could affect diesel generator interdependence," from TS SR 4.8.1.1.2.h. The licensee also proposed to update the TS Bases by adding the following statement to Section 3/4.8:

> Any time the OPERABILITY of a diesel generator has been affected by repair, maintenance, or replacement activity, or by modification that could affect its interdependency, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

In the letter dated March 28, 2006, (Ref. 1) the licensee stated that any time the operability of a system or component has been affected by repair, maintenance, or replacement, post maintenance testing is required to demonstrate the operability of the system or component in accordance with SR 4.0.1. SR 4.0.1 states that failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be a failure to meet the limiting condition for operation. Furthermore, the bases for SR 4.0.1 state that upon completion of maintenance, appropriate post maintenance testing is required to declare equipment operable.

In addition, the licensee stated that deletion of this SR from the TS will neither affect the requirements to verify operability of affected subsystems following any maintenance on or modification to these subsystems, nor will it affect the assurance that the accident analysis assumptions are satisfied.

The staff was concerned that the requirement to start both EDGs simultaneously after any modification that could affect EDG interdependence would not be preserved. In response dated January 26, 2007, to the NRC staff's concern, the licensee stated that the proposed change deletes the specific language in the SR related to post-maintenance testing but retains the balance of the SR to periodically test for EDG interdependence. Additionally, neither the method of performance nor the associated acceptance criteria for the specified tests are affected by the proposed change. As such, any maintenance activity affecting the ability of the EDGs to meet the acceptance criteria specified in the modified SR would make the previous test results invalid and thereby require the associated SR to be re-performed to verify operability.

The licensee also stated that the procedure SP 3646A.3, "Diesel Generator Independence Test," is the MPS3 procedure directing performance of SR 4.8.1.1.2.h. Section 1.4, "Frequency," specifies that this procedure is performed once every 10 years or after modifications which could affect diesel generator interdependence. Step 4.12 of that procedure requires both EDGs to be started simultaneously, consistent with the TS SR. This procedure's acceptance criteria and frequency are unaffected by the proposed change. On this basis, the NRC staff concludes that deleting this redundant SR pertaining to post-modification testing of the EDGs is acceptable.

3.6 SUMMARY

The staff has reviewed the licensee's proposed TS changes and supporting documentation. Based on its review discussed above, the NRC staff determined that even with the proposed TS deletions of SR 4.1.3.4.b, SR 4.6.6.1.b, SR 4.7.7.c, SR 4.7.9.b, a portion of SR 4.5.2.g.1, SR 4.5.2.h and SR 4.8.1.1.2.h for MPS3, 10 CFR 50.36 continues to be met and, therefore, these changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The Connecticut State official agreed with the NRC staff's conclusion as stated in Section 6.0 of this Safety Evaluation.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant change in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 29673). Accordingly, the amendment meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The NRC staff concludes that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activity will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or health and safety of the public.

7.0 REFERENCES

1. Letter from E. Grecheck (DNC) to NRC, "Dominion Power Nuclear Connecticut, Inc. Millstone Power Station Unit 3 - Proposed Revision to Technical Specifications (LBDCR 05-MP3-025) Post Maintenance or Modification Surveillance Requirements," dated March 28, 2006.

2. Letter from G. T. Bischof (DNC) to NRC, "Dominion Power Nuclear Connecticut, Inc. Millstone Power Station Unit 3 - Response to Request for Additional Information Regarding Post Maintenance/Modification Surveillance Requirements License Amendment Request," dated October 26, 2006.

3. Letter from G. T. Bischof (DNC) to NRC, "Dominion Power Nuclear Connecticut, Inc. Millstone Power Station Unit 3 - Supplement to Post Maintenance/Modification Surveillance Requirements License Amendment Request," dated December 4, 2006.

4. Letter from G. T. Bischof (DNC) to NRC, "Dominion Power Nuclear Connecticut, Inc. Millstone Power Station Unit 3 - Response to Request for Additional Information Regarding Post Maintenance/Modification Surveillance Requirements License Amendment Request," dated January 26, 2007.

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