Dominion Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, Virginia 23060
Web Address: www.dom.com

Dominion®

September 13, 2005

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555
33650

Serial No. 05-401
MPS Lic/MAE R0
Docket No. 50-423
License No. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
LICENSE AMENDMENT REQUEST (LBDCR 05-MP3-006)
TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby requests to amend Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). The enclosed license amendment request proposes to revise Technical Specification (TS) 3/4.1.3.4, "Reactivity Control Systems, Rod Drop Time," Limiting Condition For Operation (LCO) a., by reducing the temperature at which the shutdown and control rod cluster control assemblies (RCCA) drop tests are performed from "greater than or equal to 551ºF," to "greater than or equal to 500ºF." The associated TS Bases will be updated to address the proposed changes. The approval of this change will allow greater flexibility in refueling outage scheduling. This change is consistent with the temperature at which the shutdown and control RCCA drop tests are performed in Surveillance Requirement (SR) 3.1.4.3 of NUREG 1431, Standard Technical Specifications – Westinghouse Plants," published June 2004.

The proposed amendment does not involve a significant impact on public health and safety and does not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92.

The Site Operations Review Committee and the Management Safety Review Committee have reviewed and concurred with the determinations.

Attachment 1 contains description of the proposed TS change and the Significant Hazards Consideration. Attachment 2 contains the TS marked-up pages and Attachment 3 contains the retyped pages. Attachment 4 contains the marked-up pages of the TS bases for information only. MPS3 TS bases are controlled in accordance with TS Section 6.18, "Technical Specification Bases Control program."

We request issuance of this amendment no later than December 30, 2006, with the amendment to be implemented within 60 days of issuance to support spring 2007 refueling outage.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

If you have any questions or require additional information, please contact Mr. Paul R. Willoughby at (804) 273-3572

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments:
1. Evaluation of Proposed License Amendment
2. Marked-Up Pages
3. Re-typed Pages
4. Bases Marked-up Pages
5. Millstone Unit 3 Rod Drop Data

Commitments made in this letter: None.

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

Mr. G. F. Wunder
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop 08-B-1A
Rockville, MD 20852-2738

Mr. S. M. Schneider
NRC Senior Resident Inspector
Millstone Power Station

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Dominion Nuclear Connecticut, Inc. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 13th day of September, 2005.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

PROPOSED REVISION TO TECHNICAL SPECIFICATIONS (LBDCR 05-MP3-006) TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST

EVALUATION OF PROPOSED LICENSE AMENDMENT

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

PROPOSED REVISION TO TECHNICAL SPECIFICATIONS (LBDCR 05-MP3-006) TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST EVALUATION OF PROPOSED LICENSE AMENDMENT

1.0 DESCRIPTION

2.0 PROPOSED CHANGE

3.0 BACKGROUND
- 3.1 Description of The Reactivity Control System
- 3.2 Reason for Proposed Amendment

4.0 TECHNICAL ANALYSIS
- 4.1 Details of the Proposed Amendment
- 4.2 Safety Summary

5.0 REGULATORY ANALYSIS
- 5.1 No Significant Hazards Consideration
- 5.2 Applicable Regulatory Requirements/Criteria

6.0 ENVIRONMENTAL CONSIDERATION

7.0 PRECEDENTS

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby requests to amend Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). The enclosed license amendment request proposes to revise Technical Specification (TS) 3/4.1.3.4, "Reactivity Control Systems, Rod Drop Time," Limiting Condition For Operation (LCO) a., by reducing the temperature at which the shutdown and control rod cluster control assemblies (RCCA) drop tests are performed from "greater than or equal to 551 ºF," to "greater than or equal to 500ºF." The associated TS bases will be updated to address the proposed changes.

2.0 PROPOSED CHANGE

TS 3/4.1.3.4, "Reactivity Control Systems, Rod Drop Time," LCO states:

"The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating."

DNC is proposing to change the LCO, as follows:

"The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating."

3.0 BACKGROUND

3.1 Description of The Reactivity Control System

The OPERABILITY (i.e., trippability) of the shutdown and control RCCAs is an initial assumption in all safety analyses that assume RCCA insertion upon reactor trip.

There are 61 RCCAs (control rods) in the reactor core with 193 fuel assemblies. The 61 control rods are divided into five (5) shutdown banks and four (4) control banks. Shutdown banks provide the necessary reserve negative reactivity, when fully inserted, to ensure that the reactor is shutdown ($k_{eff} < 1.0$) in the event of a reactor trip. When

the reactor is operating, the control banks are used to add or remove negative reactivity to control T_{ave} by partial insertion into the core or by partial withdrawal from the core.

Verification of RCCA drop times allows the operator to determine that the maximum RCCA drop time permitted is consistent with the assumed RCCA drop time used in the safety analysis. After reactor vessel head removal and replacement during refueling outages, measuring RCCA drop times prior to reactor criticality ensures that the reactor internals and RCCA drive mechanism will not interfere with RCCA motion or RCCA drop time. It also verifies that no degradation in these systems that would adversely affect RCCA motion or drop time has occurred.

3.2 Reason for Proposed Amendment

During planning for refueling outages, DNC determined that it is possible to schedule the RCCA drop testing when the reactor coolant temperature reaches 500°F. However, TS 3.1.3.4.a currently requires that the RCCA drop test be performed at 551°F or greater. The RCCA drop testing is a critical path item, and revising the TS and its bases provides operational flexibility by permitting RCCA drop testing to be performed concurrently with other refueling outage tasks performed with reactor coolant greater than or equal to 500°F.

4.0 TECHNICAL ANALYSIS

4.1 Details of the Proposed Amendment

The RCCA drop test is intended to provide verification that the actual RCCA drop times are consistent with the RCCA drop times assumed in the safety analysis. The RCCA drop test ensures that the reactor internals and the RCCA drive mechanisms do not interfere with RCCA motion or increase the RCCA drop time, and that no degradation in the system has occurred that would adversely affect the operability of the RCCAs.

The current requirement, to perform the RCCA drop test when the average reactor coolant temperature is greater than or equal to 551ºF, ensures that the measured RCCA drop times will be representative of the conditions that exist at reactor full power operation. The value of 551ºF for RCCA drop testing is identical to the MPS3 minimum required temperature for criticality, and RCCA drop testing at 551ºF demonstrates operability at operating temperature.

During the evolution of the Westinghouse Standard Technical Specifications (NUREG-1431) an average reactor coolant temperature of greater than or equal to 500ºF was determined to adequately simulate operating conditions for RCCA drop tests. Data obtained by DNC from the MPS3 RCCA drop time testing during initial (Cycle 1) plant startup support the NUREG-1431 determination. These data demonstrate that RCCA drop times increase with decreasing reactor coolant temperature, principally because of

the increased water density and viscosity at the lower temperatures. Additionally, during the review of a Florida Power and Light license amendment request (Docket Nos. 50-250 and 50-251, submittal dated March 12, 2001) to lower the Turkey Point Units 3 and 4 RCCA drop test temperature requirements, the Nuclear Regulatory Commission (NRC) reviewed Turkey Point Units 3 and 4 initial startup RCCA drop data. The NRC's safety evaluation (Docket Nos. 50-250 and 50-251, Amendment 208, dated March 12, 2001) states that the data show that there is a slight increase in RCCA drop time as reactor coolant temperature is decreased.

Attachment 5 provides a tabulation of the RCCA drop times measured during Cycle 1 startup. The parameters for the initial tests were as follows:

- Cycle 1, 100% flow, 'cold' conditions (i.e. T_{avg} =145ºF, Reactor Coolant System (RCS) pressure = 390 psia)

- Cycle 1, 100% flow, 'hot' conditions (i.e. T_{avg} = 557ºF, RCS pressure = 2250 psia).

The increase in RCCA drop time due to decrease in temperature from 557ºF to 145ºF was 0.099 second as demonstrated by RCCA drop results in Cycle 1. This general result was expected, since the viscosity of the water was lower for the hot condition. It is assumed that the Cycle 1 result is typical of current cycles. Additionally, a factor of safety of 1.5 is applied to the increase in RCCA drop time to account for uncertainties. Therefore, the increase in RCCA drop time due to temperature decrease is conservatively estimated to be 0.15 second.

Measured RCCA drop times taken during MPS3, Cycle 10 startup were less than 1.6 seconds, and measuring the RCCA drop time at 500ºF is expected to increase this time by less than 0.15 seconds. This results in a conservative drop time estimate at 500ºF of approximately 1.75 seconds. There is sufficiently large margin between the estimated RCCA drop time and the 2.7-second limit in the TS and the 2.19 seconds for surveillance testing acceptance criteria (plant specific seismic allowance of 0.51 seconds). Therefore, there is no change in the acceptance criteria for RCCA drop time. The RCCA drop time assumption in the safety analysis is not changed, and consequently, the analysis results are not affected.

Based on the above, the available margin in the measured RCCA drop test will accommodate the slight increase in drop times as a result of performing the test at a lower temperature.

4.2 Safety Summary

DNC is proposing to change the temperature at which the shutdown and control RCCA drop tests are performed from "greater than or equal to 551ºF," to "greater than or equal

to 500ºF." This change does not alter any of the assumptions used in the safety analyses, nor will it cause any safety system parameters to exceed their acceptance limit.

The proposed change does not affect the revisions to plant procedures, which were made to address Westinghouse Nuclear Safety Advisory Letter, NSAL-00-016 (Rod Withdrawal from Subcritical Protection in Lower Modes, issued in 2000). NSAL-00-016 indicates that the core should be borated to an all-rods-out condition, when the control rod system is capable of rod withdrawal and the power range trip is not operable, to prevent core criticality from being reached during a postulated uncontrolled rod/bank withdrawal event. Revisions to plant procedures were made such that the shutdown and control banks are incapable of being withdrawn in modes 3, 4 and 5, unless:

- The Power Range Neutron Flux - low setpoint trip function is reinstated at a RCS T_{avg} greater than or equal to 551°F, or

- The core is borated to an all-rods-out condition such that criticality is precluded during a postulated uncontrolled rod/bank withdrawal accident.

Therefore, based on the above discussion, the proposed change will have no adverse effect on plant safety. Additionally, these changes can be made without adverse impact to plant operations or to the health and safety of the public.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

DNC is proposing to change the temperature at which the shutdown and control RCCA drop tests are performed from "greater than or equal to 551ºF," to "greater than or equal to 500ºF."

DNC has evaluated whether or not a Significant Hazards Consideration (SHC) is involved with the proposed changes by addressing the three standards set forth in 10 CFR 50.92(c) as discussed below.

Criterion 1:

Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

DNC is proposing to change the temperature at which the shutdown and control RCCA drop tests are performed from "greater than or equal to 551ºF," to "greater than or equal

to 500ºF." The proposed change does not modify any plant equipment and does not impact any failure modes that could lead to an accident. Additionally, the proposed change has no effect on the consequence of any analyzed accident since the change does not affect the function of any equipment credited for accident mitigation. Based on this discussion, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

Criterion 2:

Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not modify any plant equipment and there is no impact on the capability of existing equipment to perform its intended functions. No system setpoints are being modified and no changes are being made to the method in which plant operations are conducted. No new failure modes are introduced by the proposed change. The proposed amendment does not introduce accident initiators or malfunctions that would cause a new or different kind of accident.

As noted above, the proposed change does not affect the revisions to plant procedures, which were made to address Westinghouse Nuclear Safety Advisory Letter, NSAL-00-016 (Rod Withdrawal from Subcritical Protection in Lower Modes, issued in 2000).

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3:

Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The TS change does not involve a significant reduction in margin because the acceptance criterion for the RCCA drop time will not change. The proposed change will reduce the minimum RCCA drop test temperature from greater than or equal to 551ºF to greater than or equal to 500ºF. This will slightly increase the measured test RCCA drop time. However, the measured test RCCA drop time is required to remain within the current TS limit of 2.7 seconds and the 2.19 seconds for surveillance testing acceptance criteria (plant specific seismic allowance of 0.51 seconds). The proposed change does not affect any of the assumptions used in the accident analysis, nor does it affect any operability requirements for equipment important to plant safety. Therefore, the margin of safety is not impacted by the proposed amendment.

In summary, DNC concludes that the proposed amendment does not represent a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

5.2 Applicable Regulatory Requirements/Criteria

DNC is proposing to change the temperature at which the shutdown and control RCCA drop tests are performed from "greater than or equal to 551ºF," to "greater than or equal to 500ºF."

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) contains the following GDCs, which are applicable to the proposed amendment:

- GDC 10, "Reactor design," which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 26, "Reactivity control system redundancy and capability," which requires that two independent reactivity control systems of different design principles be provided. GDC 26 also requires that: (1) one of the systems shall use control RCCAs, preferably including a positive means for inserting the RCCAs, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions, such as stuck RCCAs, specified acceptable fuel design limits are not exceeded; (2) the second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded; and (3) one of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- GDC 27, "Combined reactivity control systems capability," which requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions, and with appropriate margin for stuck RCCAs, the capability to cool the core is maintained.

- GDC 28, "Reactivity limits," which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than

limited local yielding, nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. GDC 28 also requires that these postulated reactivity accidents shall include consideration of RCCA ejection (unless prevented by positive means), RCCA dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Measured RCCA drop times taken during MPS3, Cycle 10 startup were less than 1.6 seconds, and measuring the RCCA drop time at 500ºF is expected to increase this time by less than 0.15 seconds to become approximately 1.75 seconds. There is sufficient margin between the estimated RCCA drop time and the 2.7-second limit in the TSs. There is no change in the acceptance criteria for RCCA drop time. Therefore, the above listed criteria are not impacted by the proposed change.

In 10 CFR 50.36, the U.S. Nuclear Regulatory Commission (NRC or the Commission) established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, technical specifications are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's technical specifications. The proposed change ensures that Technical Specification 3/4.1.3.4, "Reactivity Control Systems, Rod Drop Time," Limiting Condition For Operation (LCO) a., will continue to satisfy the requirements of 10 CFR 50.36.

6.0 ENVIRONMENTAL CONSIDERATION

DNC has determined that the proposed amendment would change requirements with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, or it would change inspection or surveillance requirements. DNC has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released off site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENTS

A Florida Power and Light license amendment request (Docket Nos. 50-250 and 50-251) was submitted on March 12, 2001 to lower the Turkey Point Unit 3 and 4 RCCA

drop test temperature requirements. The NRC approved Florida Power and Light license amendment request on May 7, 2001 (Amendments 214 and 208).

A similar request was made by Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Unit 1 (Docket No. 50-315) on June 25, 2004. The NRC approved this request on December 20, 2004 (Amendment 284).

ATTACHMENT 2

**PROPOSED REVISION TO TECHNICAL SPECIFICATIONS (LBDCR 05-MP3-006)
TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST**

MARKED-UP PAGE

**MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.**

~~July 24, 2002~~

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to ~~551~~ 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

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 those specific rods, and

c. At least once per 24 months.

ATTACHMENT 3

**PROPOSED REVISION TO TECHNICAL SPECIFICATIONS (LBDCR 05-MP3-006)
TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST**

RE-TYPED PAGE

**MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.**

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

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 those specific rods, and

c. At least once per 24 months.

ATTACHMENT 4

PROPOSED REVISION TO TECHNICAL SPECIFICATIONS (LBDCR 05-MP3-006) TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST

BASES MARKED-UP PAGES

MILLSTONE POWER STATION UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control

MOVABLE CONTROL ASSEMBLIES (Continued)

rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120, and fully withdrawn position for the Control Banks and 18, 210, and fully withdrawn position for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to ~~551~~ 500°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

The required rod drop time of ≤ 2.7 seconds specified in Technical Specification 3.1.3.4 is used in the FSAR accident analysis. A rod drop time was calculated to validate the Technical Specification limit. This calculation accounted for all uncertainties, including a plant specific seismic allowance of 0.51 seconds. Since the seismic allowance should be removed when verifying the actual rod drop time, the acceptance criteria for surveillance testing is 2.19 seconds (References 4 and 5).

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The Digital Rod Position Indication (DRPI) System is defined as follows:

- Rod position indication as displayed on DRPI display panel (MB4), or
- Rod position indication as displayed by the Plant Process Computer System

With the above definition, LCO, 3.1.3.2, "ACTION a." is <u>not</u> applicable with either DRPI display panel or the plant process computer points OPERABLE.

The plant process computer may be utilized to satisfy DRPI System requirements which meets LCO 3.1.3.2, in requiring diversity for determining digital rod position indication.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable,

For Information Only

MOVABLE CONTROL ASSEMBLIES (Continued)

then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.

Additional surveillance is required to ensure the plant process computer indications are in agreement with those displayed on the DRPI. This additional SURVEILLANCE REQUIREMENT is as follows:

> Each rod position indication as displayed by the plant process computer shall be determined to be OPERABLE by verifying the rod position indication as displayed on the DRPI display panel agrees with the rod position indication as displayed by the plant process computer at least once per 12 hours.

The rod position indication, as displayed by DRPI display panel (MB4), is a non-QA system, calibrated on a refueling interval, and used to implement T/S 3.1.3.2. Because the plant process computer receives field data from the same source as the DRPI System (MB4), and is also calibrated on a refueling interval, it fully meets all requirements specified in T/S 3.1.3.2 for rod position. Additionally, the plant process computer provides the same type and level of accuracy as the DRPI System (MB4). The plant process computer does not provide any alarm or rod position deviation monitoring as does DRPI display panel (MB4).

For Information Only

MOVABLE CONTROL ASSEMBLIES (Continued)

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

For LCO 3.1.3.6 the control bank insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR). These insertion limits are the initial assumptions in safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions, assumptions of available SHUTDOWN MARGIN, and initial reactivity insertion rate.

The applicable I&C calibration procedure (Reference 1.) being current indicates the associated circuitry is OPERABLE.

There are conditions when the Lo-Lo and Lo alarms of the RIL Monitor are limited below the RIL specified in the COLR. The RIL Monitor remains OPERABLE because the lead control rod bank still has the Lo and Lo-Lo alarms greater than or equal to the RIL.

When rods are at the top of the core, the Lo-Lo alarm is limited below the RIL to prevent spurious alarms. The RIL is equal to the Lo-Lo alarm until the adjustable upper limit setpoint on the RIL Monitor is reached, then the alarm remains at the adjustable upper limit setpoint. When the RIL is in the region above the adjustable upper limit setpoint, the Lo-Lo alarm is below the RIL.

References:

1. IC 3469N08, Rod Control Speed, Insertion Limit, and Control TAVE Auctioneered/Deviation Alarms.

2. Letter NS-OPLS-OPL-1-91-226, (Westinghouse Letter NEU-91-563), dated April 24, 1991.

3. Millstone Unit 3 Technical Requirements Manual, Appendix 8.1, "CORE OPERATING LIMITS REPORT".

4. Westinghouse Letter NEU-97-298, "Millstone Unit 3 - RCCA Drop Time," dated November 13, 1997.

5. Westinghouse Letter 98NEU-G-0060, "Millstone Unit 3 - Robust Fuel Assembly (Design Report) and Generic SECL," dated October 2, 1998.

ATTACHMENT 5

**PROPOSED REVISION TO TECHNICAL SPECIFICATIONS (LBDCR 05-MP3-006)
TEMPERATURE REQUIREMENT FOR THE REACTIVITY CONTROL SYSTEM ROD DROP TIME TEST**

MILLSTONE UNIT 3 ROD DROP DATA

**MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.**

Millstone Unit 3 RCCA Drop Time Data

Rod Bank	Core Location	Cycle 1 : Initial core/ Drop Time Testing 100% RCS Flow Cold Conditions Drop Time (msec)	100% RCS Flow Hot Conditions Drop Time (msec)	Difference in Drop Times C1(cold) - C1(hot) (msec)
SB-A	D02	1500	1412	88
	B12	1492	1402	90
	M14	1506	1416	90
	P04	1508	1422	86
	H04	1514	1404	110
	B04	1492	1274	218
	D14	1488	1398	90
	P12	1496	1394	102
	M02	1494	1418	76
	H12	1508	1408	100
SB-B	G03	1486	1396	90
	C09	1498	1400	98
	J13	1480	1376	104
	N07	1500	1416	84
	D08	1496	1410	86
	C07	1492	1400	92
	G13	1504	1398	106
	N09	1498	1402	96
	J03	1494	1398	96
	M08	1494	1406	88
SB-C	E03	1492	1398	94
	C11	1512	1396	116
	L13	1502	1388	114
	N05	1512	1398	114
SB-D	C05	1476	1394	82
	E13	1496	1402	94
	N11	1498	1402	96
	L03	1494	1392	102
SB-E	A07	1486	1398	88
	G15	1494	1402	92
	R09	1498	1396	102
	J01	1480	1406	74

Cycle 1 : Initial core

Rod Bank	Core Location	Drop Time Testing 100% RCS Flow Cold Conditions Drop Time (msec)	100% RCS Flow Hot Conditions Drop Time (msec)	Difference in Drop Times C1(cold) - C1(hot) (msec)
CB-A	H06	1558	1406	152
	F08	1542	1386	156
	H10	1488	1364	124
	K08	1494	1408	86
	E05	1510	1392	118
	E11	1508	1410	98
	L11	1498	1400	98
	L05	1498	1366	132
CB-B	F02	1504	1420	84
	B10	1490	1418	72
	K14	1506	1422	84
	P06	1500	1410	90
	B06	1480	1400	80
	F14	1482	1398	84
	P10	1496	1408	88
	K02	1482	1386	96
CB-C	H02	1488	1420	68
	B08	1498	1400	98
	H14	1498	1398	100
	P08	1519	1396	123
	F06	1490	1402	88
	F10	1496	1404	92
	K10	1492	1392	100
	K06	1492	1402	90
CB-D	D04	1480	1392	88
	M12	1482	1406	76
	D12	1488	1382	106
	M04	1498	1402	96
	H08	1508	1398	110
Average Times		**1497**	**1399**	**99**