August 27, 2001

Mr. R. P. Necci
Vice President - Nuclear Technical Services
c/o Mr. David A. Smith
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN CURVES (TAC NO. MB1785)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3, in response to your application dated April 23, 2001, as supplemented by letters dated June 25, June 29, and July 19, 2001.

The amendment modifies Technical Specifications (TSs) to relocate the boration subsystem and Residual Heat Removal System over-pressurization protection requirements (Modes 4 and 5) to a licensee controlled document; to revise the Reactor Coolant System pressure/temperature limits; to revise the Cold Overpressure Protection System setpoint curves, enable temperatures and associated restrictions; to revise the reactor vessel material surveillance withdrawal schedule; to revise the pressurizer code safety valve requirements; and the Index and the associated Bases for these TSs will be modified as a result of the changes.

On a separate but related note, the Commission has made it known of the need for effective and efficient use of staff resources. The Commission considers the non-timely submittal of this application (April 23, 2001, with a licensee need date of August 30, 2001, creating an unacceptably short staff review time), complicated by inclusion of changes to non-related TS items does not contribute to the Commission achieving its goal for effective and efficient use of staff resources. I discussed this with Mr. David A. Smith of your staff and anticipate that he will give this the attention it deserves.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Victor Nerses, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 197 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Victor Nerses, Sr. Project Manager, Section 2
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DOMINION NUCLEAR CONNECTICUT, INC., ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. NPF-49

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

   A. The application for amendment by the applicant dated April 23, 2001, as supplemented June 25, June 29, and July 19, 2001, 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 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;

   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. 197 , 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 the date of issuance, and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

EAdensam for /RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 27, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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

Remove	Insert
iv	iv
vii	vii
xiii	xii
3/4 1-13	3/4 1-13
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
3/4 1-17	3/4 1-17
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 4-2	3/4 4-2
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
3/4 4-5a	3/4 4-5a
3/4 4-6	3/4 4-6
3/4 4-6a	3/4 4-6a
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-38	3/4 4-38
3/4 4-38a	3/4 4-38a
3/4 4-39	3/4 4-39
3/4 4-40	3/4 4-40
3/4 4-41	3/4 4/41
B3/4 1-2	B3/4 1-2
B3/4 1-3	B3/4 1-3
B3/4 1-3a	- - -
B3/4 4-1	B3/4 4-1
B3/4 4-1a	B3/4 4-1a
B3/4 4-2	B3/4 4-2
B3/4 4-7	B3/4 4-7
B3/4 4-8	B3/4 4-8

Remove	Insert
B3/4 4-9	B3/4 4-9
B3/4 4-10	B3/4 4-10
B3/4 4-11	B3/4 4-11
B3/4 4-12	B3/4 4-12
B3/4 4-15	
B3/4 4-16	B3/4 4-16
B3/4 4-16a	B3/4 4-16a
B3/4 4-17	B3/4 4-17
B3/4 4-18	B3/4 4-18
B3/4 4-19	B3/4 4-19
B3/4 4-20	B3/4 4-20
B3/4 4-21	B3/4 4-21
B3/4 4-22	B3/4 4-22
B3/4 4-23	B3/4 4-23
B3/4 4-24	B3/4 4-24
B3/4 4-25	B3/4 4-25
B3/4 4-26	B3/4 4-26

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197

TO FACILITY OPERATING LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated April 23, 2001, as supplemented by letters dated June 25, June 29, and July 19, 2001, Dominion Nuclear Connecticut, Inc., (DNC/licensee) submitted a request to revise the Technical Specifications (TSs) for the Millstone Nuclear Power Station, Unit No. 3 (MP3). Specifically, the licensee requested approval to amend the TS to revise the pressure-temperature (PT) curves, the cold overpressure protection limits (COPPS) and approval for an exemption to use the American Society of Mechanical Engineers (ASME) Code Case N-640 to satisfy the requirements of Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. The request for an exemption and use of the ASME Code case N-640 was reviewed concurrently with this amendment but as a separate action. The granting of the exemption and the use of ASME Code Case N-640 was documented in NRC letter dated August 14, 2001 (Accession No.: ML012010129).

The amendment also removes and relocates to the technical requirements manual (TRM) the TS requirements for boration in the reactivity control system. In addition, the licensee proposed modifications to the index and the associated Bases for a number of these TSs. It should be noted that a review of the proposed revisions to TS 3.4.1.6 is not included in this safety evaluation and will be completed at a later date. The letters dated June 25, June 29, and July 19, 2001, provided clarifying information and did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the application as published in the *Federal Register*.

2.0 BACKGROUND

2.1 Integrity of Reactor Coolant Pressure Boundary

The Nuclear Regulatory Commission (NRC) has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requests that licensees submit their reactor pressure vessel (RPV) data to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code.

The ASME Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements by applying an alternate reference fracture toughness for reactor vessel materials in determining P-T limits, as permitted by Code Case N-640, while maintaining, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety. The granting of the exemption and the use of ASME Code Case N-640 was documented as noted above.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 thickness (1/4T) of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

2.2 Relocation of Technical Specifications

The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. This regulation requires that the TSs include items in five specific categories. These categories include 1) safety limits, limiting safety system settings and limiting control settings, 2) limiting conditions for operation, 3) surveillance requirements, 4) design features, and 5) administrative controls. However, the regulation does not specify the particular TSs to be included in a plant's license.

Additionally, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a limiting condition for operation (LCO) is required to be included in the TSs. These criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Existing LCOs and related surveillances included as TS requirements which satisfy any of the criteria stated above must be retained in the TSs. Those TS requirements which do not satisfy these criteria may be relocated to other licensee-controlled documents.

The effect of the proposed change is the following: relocate TSs 3.1.2.1 through 3.1.2.6 to the MP3 TRM, and revise TSs 3.4.1.2 and 3.4.1.3. The MP3 TS Bases will also be modified as a result of the proposed changes.

3.0 EVALUATION

3.1 Pressure-Temperature Limits

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the reactor vessel of MP3. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The material with the highest ART at 32 EFPY is the intermediate shell B9805-1, with 0.05% copper (Cu), 0.64% nickel (Ni), and an initial RT_{NDT} of 60 °F. Although the information from the Capsule X surveillance report (May 2000) indicated that the licensee could use the chemistry factor based on Section 2.1 (surveillance data) of

RG 1.99, Rev. 2, for the limiting material, the licensee determined to use the more conservative chemistry factor based on the chemistry factor table in Section 1.1 of the RG in its P-T limit calculations. Using this conservative approach, the staff calculated the ARTs to be 124.8 °F at 1/4T and 107.0 °F at 3/4T for the limiting material. These ARTs were calculated at 32 EFPY with a surface neutron fluence of 1.97E19 n/cm^2. The licensee's ARTs of 123.6 °F at 1/4T and 105.8 °F at 3/4T are very close to the staff's values. Both the staff and the licensee did not include the cladding thickness in calculating the attenuation of the fluence through the vessel wall.

The ARTs calculated by the staff and the licensee are almost identical. However, substituting the ART of 123.6 °F into equations in Appendix G of the 1995 ASME Code, the staff found big discrepancies between the proposed P-T limits and those calculated by the staff. In the response dated June 26, 2001, to the staff's request for additional information, the licensee confirmed the following plant-special features in the proposed P-T limits for Millstone 3: (1) indicator uncertainties, 25.3 °F for temperature and 115.5 psia for pressure; (2) a pressure drop of 28.3 psi between the pressure transmitter and the reactor vessel beltline for one pump operation and a pressure drop of 74 psi for four pump operation; (3) composite curves derived from the most restrictive pressure value considering the specified rates and isothermal conditions; and (4) an adjustment of 10 psi to the gage pressure to account for the primary containment pressure. After considering these extra conservatisms, the staff confirmed that the proposed P-T limits (for 32 EFPY) for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50 as modified by Code Case N-640. Code Case N-640 permits the use of an alternate reference fracture toughness for reactor vessel materials in determining P-T limits while maintaining, pursuant to 10 CFR 50.12(a)(2)(i), the underlying purposes of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

In addition to beltline materials, Appendix G of 10 CFR Part 50 imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the limiting nozzle shell reference temperatures of 40 °F, the minimum allowable temperature of this region is 160 °F. As mentioned previously, to account for indicator uncertainty, the licensee added 25.3 °F to 160 °F to obtain 185.3 °F. These limits are shown as straight-line segments on the proposed P-T limits of the submittal. The indent from 160 °F to 185.3 °F was caused by switching from one reactor coolant pump (RCP) operation to four RCP operation (see (2) in the previous paragraph.) In summary, the staff has determined that the proposed P-T limits satisfy the requirements in Section IV.A.2 of Appendix G.

Appendix G further requires that the predicted Charpy USE at end-of-license (EOL) for vessel beltline materials be above 50 ft-lb or that licensees demonstrate that lower values of Charpy USE will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code. This USE requirement is satisfied because all beltline materials have EOL USEs above 50 ft-lb.

The submittal also contains the bases and the revised LCO related to P-T limits and a revised RPV surveillance capsule withdrawal schedule in the TSs. The revised bases and the LCO are acceptable because the changes are either non-technical or consistent with the revised P-T analysis. The revised capsule withdrawal schedule is acceptable because it meets the requirements of ASTM E185-82.

In summary, the staff considers that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality satisfy the requirements of Appendix G to Section XI of the ASME Code, as modified by Code Case N-640, and Appendix G of 10 CFR Part 50 for 32 effective full power years (EFPY). The proposed P-T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Hence, the proposed P-T limits may be incorporated into the MP3 Technical Specifications. The proposed changes to the bases and the limiting condition for operation related to the P-T limits and to the surveillance capsule withdrawal schedule in the TS are also acceptable.

3.2 Technical Specifications Changes

3.2.1 Relocation of TSs 3.1.2.1 through 3.1.2.6 to TRM

The existing conditions, actions, and surveillance requirements for TSs 3.1.2.1 through 3.1.2.6, which deal with maintaining shutdown margin (SDM) by boration, will be relocated to the TRM. This group of TSs addresses the boration subsystem of the Chemical and Volume Control System. These requirements define the minimum operable boron injection flow paths, the minimum operable charging pumps in the boron injection path, and the minimum operable borated water sources. Operation of the boration subsystem is not credited for mitigation of any design basis accident or transient. The relocation of TSs 3.1.2.1 through 3.1.2.6 only deals with the boration function of the affected equipment (i.e., charging/high head safety injection pumps and the refueling water storage tank) to maintain SDM not the emergency core cooling aspects of the same equipment or the beyond design basis Anticipated Transient Without Scram (ATWS) events. There are no proposed changes to the TSs for the emergency core cooling or ATWS aspects of the affected equipment. Thus, this evaluation of the relocation of TSs 3.1.2.1 through 3.1.2.6 only considered the boration function of the affected equipment to maintain SDM.

The staff evaluated the existing TSs against the four criteria set forth in 10 CFR 50.36(c)(2)(ii). The boration subsystem and borated water sources are not a form of instrumentation or a process variable, design feature or operating restriction that is an initial condition of a design basis accident or transient, and therefore, do not meet Criteria 1 and 2. The boration subsystem and borated water sources are a structure, system, or component. However, the boration function of the system to maintain SDM is not a primary success path that functions or actuates to mitigate a design basis accident or transient. The accident analyses assume that the required SDM at the start of an accident has been established since the TS SDM requirements have to be met before entering the plant's Mode of Applicability. This provides sufficient time for the plant operators to recognize the event and terminate the event prior to a complete loss of SDM. Therefore, the boration subsystem and borated water sources that function to maintain SDM do not meet Criterion 3. The boration subsystem and borated water sources are a low risk contribution in the MP3 Individual Plant Examination; therefore, the risk to public health and safety is not significant. Therefore, the boration subsystem and borated water sources do not meet Criterion 4 for inclusion in TSs. Since TSs 3.1.2.1 through 3.1.2.6 requirements do not satisfy these criteria, TSs 3.1.2.1 through 3.1.2.6 may be relocated to other licensee-controlled documents.

The licensee has incorporated the TRM into the MP3 Final Safety Analysis Report. As such, changes to the TRM would be controlled in accordance with approved station procedures and the requirements of 10 CFR 50.59. Therefore, the staff considers that sufficient regulatory controls exist and concludes that TSs 3.1.2.1 through 3.1.2.6 may be relocated from the TSs to the licensee's TRM.

3.2.2 TS 3.4.1.2

TS LCO 3.4.1.2 provides the requirements for the reactor coolant system (RCS) in MODE 3. Specifically, LCO 3.4.1.2 requires at least three reactor coolant loops be OPERABLE, with at least three reactor coolant loops in operation when the Reactor Trip System breakers are closed or with at least one reactor coolant loop in operation when the Reactor Trip System breakers are open. The licensee proposed to revise the wording in LCO 3.4.1.2 by replacing the phrases "Reactor Trip System breakers are closed" and "Reactor Trip System breakers are open" with the phrases "Control Rod Drive System is capable of rod withdrawal" and "Control Rod Drive System is not capable of rod withdrawal," respectively. The current TS only allows one method to prevent control rod withdrawal, that is, open the reactor trip breakers. The result of proposed changes would provide additional operational flexibility since alternate methods of preventing control rod withdrawal can be used. Such methods include opening the 480 VAC input breakers to the control rod drive motor generator sets and the control rod drive motor generator sets output breakers. The proposed changes do not alter the reactor coolant system RCS loop and flow requirements. The proposed changes allow more than one method that can be used to prevent control rod withdrawal, and this flexibility should allow operators more opportunity to focus on issues important to safety. The resultant LCO maintains an adequate degree of protection consistent with the safety analysis. It also improves focus on issues important to safety and provides reasonable operational flexibility without adversely affecting the safe operation of the plant.

The licensee also proposed to revise the wording in Action b. The proposed change would replace the phrase "Reactor Trip System breakers in the closed position" with "Control Rod Drive System is capable of rod withdrawal." As discussed above, the proposed change would provide operational flexibility and does not alter the RCS loop or flow requirements.

As discussed above, the staff concludes that the proposed changes to TS LCO 3.4.1.2 and Action b are acceptable. Although not mentioned in the license amendment request, the proposed changes also align TS LCO 3.4.1.2 and Action b to the APPLICABILITY of TS 3.3.1, Tables 3.3-1 and 4.3-1. The proposed changes are also consistent with NUREG-1431, Standard Technical Specifications Westinghouse Plants.

3.2.3 TS 3.4.1.3

TS LCO 3.4.1.3 provides the requirements for the reactor coolant system in MODE 4. Specifically, LCO 3.4.1.3 requires that when the Reactor Trip System breakers are closed, at least two RCS loops shall be OPERABLE and in operation, or when the Reactor Trip System breakers are open, at least two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and at least one of these loops shall be in operation. The licensee has proposed to replace the phrases "Reactor Trip System breakers closed" and "Reactor Trip System breakers open" with the phrases "the Control Rod Drive System capable of rod withdrawal" and "the Control Rod Drive System not capable of rod withdrawal," respectively. The current TS only allows one method to prevent control rod withdrawal, that is, open the reactor trip breakers. The result of proposed changes would provide additional operational flexibility since alternate methods of preventing control rod withdrawal can be used. Such methods include opening the 480 VAC input breakers to the control rod drive motor generator sets and the control rod drive motor generator sets' output breakers. The proposed changes do not alter the reactor coolant system (RCS) loop and flow requirements. The proposed changes allow more than one method that can be used to prevent control rod withdrawal, and this flexibility should allow operators more opportunity to focus on issues important to safety. The resultant LCO maintains an adequate degree of protection consistent with the safety analysis. It also improves focus on issues important to safety and provides reasonable operational flexibility without adversely affecting the safe operation of the plant. As discussed above, the staff concludes that the proposed changes to TS LCO 3.4.1.3 are acceptable. Also, the proposed changes are also consistent with NUREG-1431.

The licensee proposed to add an additional action statement to TS 3.4.1.3. The current action requirements of TS 3.4.1.3 do not contain a required action to address less than two RCS loops in operation when the Control Rod Drive System is capable of withdrawal. The proposed action statement requires the Reactor Trip System breakers to be open within 1 hour when less than the required loops are in operation and the Control Rod Drive System is capable of rod withdrawal. The proposed action statement is considered to be a more restrictive change that will provide an additional action with less than two RCS loops in operation. Based on this change, the current Action b will become Action c in TS 3.4.1.3. The staff has reviewed the addition of the new action statement and concludes, based on the above information, that the proposed change is acceptable.

The licensee proposed to revise the reactor coolant pump (RCP) start criteria depicted as footnote ** in TS 3.4.1.3. The current criteria state that an RCP shall not be started unless one of the following conditions is met:

a. At least one RCP is operating.

b. The secondary side water temperature of each steam generator, not isolated from the RCS, is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.

c. With a maximum of one RCS loop isolated and with the RHR relief valves isolated from the RCS, the secondary side water temperature of each steam generator, not isolated from the RCS, is less than or equal to 250 °F.

d. All RCS wide range cold leg temperatures > 275 °F and no cold overpressure protection relief valves are in service as follows:

    1) COPPS is blocked or the block valves to the pressure operated relief valves (PORVs) are closed, and

    2) RHR relief valves are isolated from the RCS (3RHS*MV8701C or 3RHS*MV8701A is closed and 3RHS*MV8702B or 3RHS*MV8702C is closed).

The licensee proposed to revise the start criteria such that it only applies to the start of the first RCP (Criterion ‘a’ above). Criteria ‘b’ and ‘c’ are also being revised in the proposed criteria. The revised restrictions ensure that the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressure event beyond the capability of the COPPS. These proposed restrictions to the Pressure/Temperature Limits are addressed later in this Evaluation. The revised RCP starting criteria are based on the equipment used to provide cold overpressure protection. A maximum temperature differential of 50 °F between the steam generator secondary sides and RCS cold legs will limit the potential energy addition to within the capability of the PORV to mitigate the transient. The RHR relief valves are also adequate to mitigate energy addition transients constrained by this temperature differential limit, provided all RCS cold leg temperatures are at or below 150 °F. The revised RCP start criteria will state:

> The first reactor coolant pump shall not be started when any RCS loop wide range cold leg temperature is ≤ 226 ° F unless:
>
> a. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature; OR
>
> b. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

RCP starting Criterion ‘d’ and the restrictions on mass input prior to placing the RHR system into service (footnote ***) are designed to prevent overpressurization of the RHR system. The licensee has proposed to remove Criterion ‘d’ and footnote *** from TS 3.4.1.3 since Criterion ‘d’ and footnote *** are associated with the protection of the RHR system. The licensee has concluded that the prevention of RHR system overpressurization due to starting an RCP or due to excessive mass addition does not meet any of the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in the TSs. The staff evaluated Criterion ‘d’ and footnote *** against the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Criterion ‘d’ and footnote *** are not forms of instrumentation that are used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Criterion ‘d’ and footnote *** are not structures, systems, or components that are part of the primary success path no do they function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Criterion ‘d’ and footnote *** are not structures, systems or components that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Therefore, criteria ‘d’ and footnote *** do not meet Criteria 1, 3, or 4. Criterion ‘d’ and footnote *** are design features which prevent overpressurization of the RHR system. These are important restrictions for the protection of the RHR system (design pressure of 600 psig), but they do not challenge the integrity of a fission product barrier. The RHR system can be isolated by closure of the inlet and outlet motor operated valves, which are designed for full RCS pressure.

Criterion ‘d’ and footnote *** are not initial conditions of a design basis accident or transient analysis. Therefore, Criterion ‘d’ and footnote *** do not meet Criterion 2 for inclusion in the TSs and may be deleted from the MP3 TSs.

The staff has reviewed the proposed changes to TS 3.4.1.3, as described above, and has concluded that the proposed changes are acceptable.

3.2.4 <u>TS 3.4.1.4.1</u>

The RCS pump starting criteria are adjusted to accommodate the results of the new PT and COPPS limit calculations. The proposed change is applicable for Mode 5 with at least one residual heat removal (RHR) system pump in operation, one operable or at least two steam generators at secondary level greater than 17 percent and at least two reactor coolant loops filled. The proposed changes are:

- Criterion “a” will be deleted. Criterion “a” prevents any coolant pump from operating with coolant temperatures less than 160 ° F, unless the COPPS is blocked or unless the PORV block valves are closed. This criterion protects the RCP seals from pressure undershoots following actuation of the PORVs by the COPPS. However, the revised analysis has shown that seal integrity will not be compromised due to revising the COPPS setpoints resulting from the use of ASME Code case N-640.

- Criterion “b1" will be retained without the restriction of operating one RCP below 160 ° F. This restriction will be relocated to the LCO of the revised TS 3.4.9.1 with the value resulting from the revised cold overpressure analysis.

- Criterion “b2" will be retained. The revised criterion will control the temperature differential between the steam generator secondary and the RCS primary when the plant is in Mode 5 to address energy input from the secondary. In this case only the RHR system relief valves can provide cold overpressure protection which is stated in the limiting condition of operation (LCO) of TS 3.4.9.3. The proposed criterion is consistent with the revised COPPS analysis.

- Criterion “b3" will be retained. This criterion addresses energy input from the secondary system with one RCP isolated and the RHR system not in service when the RCS is in Mode 5. This criterion is consistent with the revised COPPS analysis.

- Criterion “b4" will be retained. This criterion addresses energy input from the secondary system (with at most one RCS loop isolated and the RHR in service) by controlling the temperature differential between the RCS primary (cold leg temperature less than the COPPS enable temperature) and the steam generator secondary. The proposed criterion is consistent with the revised COPPS analysis in TS 3.4.9.3.

SR 4.4.1.4.1.3 requiring determination of RHR pump operability once per 7 days, is added to verify that the standby RHR pump is available.

The above changes reflect the results of the revised calculations, provide the same level of protection against overpressurization and pressure undershoot as the previous TS and agree with the provisions of the standard TS (STS). The revised calculations were based on WCAP-14040-NP-A (Ref. 5). Therefore, the staff finds them acceptable.

3.2.5 TS 3.4.1.4.2

TS LCO 3.4.1.4.2 is applicable in Mode 5 with less than two reactor coolant loops filled, one RHR loop in operation and two RHR loops operable. The following changes are proposed:

- Criterion "a" will be deleted. This criterion was intended to protect the RCP seals from vessel pressure undershoot, however, the new analysis (using ASME Code case N-640) has raised the COPPS setpoints and does not challenge the seal undershoot pressure.

- Criterion "b1" will be retained, however, the restriction of only operating below 160° F will not be retained in this TS but will be contained in TS 3.4.9.1.

- Criterion "b2" concerns energy input from the secondary side of the steam generators and will be retained. In the configuration assumed in this TS only the RHR relief valves provide overpressurization protection per TS 3.4.9.3. The revised criteria control the temperature differential between the RCS water and the secondary side of the steam generators to avoid excess energy input to the pressure vessel and thus, avoid overpressurization.

- Criterion "b3" will be retained; it addresses energy input from the secondary side with at most one RCS loop isolated and the RHR system out of service. To avoid overpressurization when the cold leg is below the COPPS temperature, the temperature differential is required to be lower than 50°F.

- Criterion "b4" will be retained. This criterion addresses energy input from the secondary side with one RCS loop isolated and the RHR system in service by requiring that the temperature of the secondary side of the steam generators be less than 50° F from the cold leg temperature of any of the RCS loops.

SR 4.4.1.4.2.1 requiring determination of RHR pump operability once per 7 days, is added to verify that the standby RHR pump is available.

The staff concludes that for the assumed plant conditions (two RHR loops operable and at least one RHR loop in operation) TS 3.4.1.4.2 includes all of the possible variations in equipment and is equivalent in scope to the existing TS and the associated LCOs for an RCP start and thus, is acceptable. In addition it complies with the corresponding STSs.

3.2.6 TS 3.4.2.1 and 3.4.2.2

The licensee proposed to combine TSs 3.4.2.1 for pressurizer code safety valves (Modes 1 through 3) and 3.4.2.2 pressurizer code safety valves (Mode 4) into TS 3.4.2 applicable in Modes 1 through 4.

The proposed change to TS 3.4.2 is more restrictive than the two separate TSs 3.4.2.1 and 3.4.2.2. This is because the new TS requires that all (versus some) code safety valves be operable in Modes 1, 2, 3 and 4. Also, the applicability range is lower in that the new TS is applicable in Modes 1, 2, 3, and 4 for temperatures above 226 °F, where 226 °F is the COPPS enable temperature. This is reasonable because the code safety valves are not suitable for cold overpressurization protection. The current action requirement is to suspend any activity which would result in positive reactivity change. In these Modes of operation suspending reactivity addition neither prevents nor mitigates an overpressurization event, while it prevents cooldown to below 226 °F to enable COPPS. The current action requirement to be in Mode 4 within 6 hours will be replaced with the action requirement to be in Mode 4 within 24 hours and be cooled down to enter TS 3.4.9.3. The staff finds this acceptable because TS 3.4.9.3 is within the Mode 4 temperature limits, and to enter TS 3.4.9.3 from Modes 1-3, the plant needs time of the order of 24 hours to cooldown. It is further noted that the plant must enter TS 3.4.9.3 for cold leg temperature less than or equal to 226 °F. Also, the staff notes that proposed changes are consistent with the STSs.

The surveillance requirements for TSs 3.4.2.1 and 3.4.2.2 will be combined This is a necessary circumstance of merging the TSs and not a technical change and it is also consistent with the STSs and, thus, the proposed change is acceptable.

3.2.7 TS 3.4.9.1 PT Limits

There are four LCOs in the existing TS. The licensee proposed to replace these with a single statement with the limitation that one RCP can be in operation when the lowest cold leg temperature is less or equal to 160 °F.

The new statement includes the LCO requirements of items “a” and “b”. The restriction of no RCP operation at or below 120 °F has been deleted based on the new COPPS analysis. Item “c” which addresses steady state operation is deleted because the content is specified in the applicability of Figures 3.4-2 and 3.4-3. Item “d” is also deleted because the requirement is contained in the figures. The new LCO specifies that the TS is applicable to ferritic materials only. Figures 3.4-2 and 3.4-3 are to be replaced with revised PT curves.

The new action statements are separated by operating Mode. For Modes 1 through 4 the 30-minute period for limit restoration, the 72-hour period for the performance of engineering evaluation and the 500 psia requirement for pressure reduction remain the same. For Modes other than 1 through 4 immediate action is required to restore the temperature and/or pressure to within limits. The changes are consistent with the STSs.

The staff finds that the proposed TS 3.4.2 is comprehensive because it includes equivalent limitations and action statements to restore required pressure-temperature conditions or enter Mode 5. The new TS is also consistent with the STSs. For the reasons given above, the staff finds the proposed TS to be acceptable.

3.2.8 TS 3.4.9.3, Overpressure Protection System

The proposed changes reflect the results of the revised COPPS analysis and bring the wording closer to the STSs. The proposed changes are:

- In items 1 and 3 the sentence: "with no more than one isolated RCS loop" is added to clarify that the pressurizer PORVs cannot be used for cold overpressure protection if two or more RCS loops are isolated.

- In several places the value of the vent size (which physically is 5.4 $inch^2$) is stated as greater than 2.0 $inch^2$ to conform with the corresponding statement in the STSs.

- The applicability is clarified to be: Mode 4 (with any cold leg wide range temperature less or equal to 226 $^\circ$F which is the COPPS enable temperature) and Modes 5 and 6. This conforms with the results of the analysis and the COPPS functional requirements.

- Increase of the time required to establish an RCS vent from 8 hours to 12 hours to be consistent with NUREG-1431. In addition, the time increase does not result in any adverse plant condition and is also consistent with the STS requirements.

The proposed changes in TS 3.4.9.3 update the specification to match the format and wording of the STSs and to incorporate the results of the revised analysis, therefore, the staff finds the proposed TS 3.4.9.3 to be acceptable.

3.3 Calculation Methodology for COPPS and LTOP Settings

The proposed revisions in the PT curves and the COPPS limits were based on the measurement and calculation results of surveillance capsule X removed from MP3 during the cycle 6 refueling outage in 1999.

The results were documented in WCAP-15405 (Ref. 1). The licensee had the dosimetry and fluence analysis performed using the DORT (Ref. 2) code with the BUGLE 96 (Ref. 3) cross sections which are based on the ENDF/B-VI data set. The licensee stated that the analysis complies with the provisions of DG-1053 (currently RG 1.190, (Ref. 4)). Although Westinghouse applied the FERRET code, which has not been reviewed by the NRC, in processing the measured data, the differences between measurement and calculation are within acceptable limits because the differences are within the limits stated in RG 1.190. Since, the differences are within the limits, the staff finds the proposed fluence value acceptable for the revision of the PT curves and the COPPS limits.

The analysis incorporates ASME Code Case N-640 to provide greater flexibility in the PORV low and high pressure settings. This results in a lower probability of challenge to the PORVs, higher margin to maintain the net positive suction head and eliminates the possibility of RCP seal damage due to pressure undershoot.

The measurements and calculations of the capsule X dosimetry predict the estimated end of license (32 EFPYs) peak inside surface fluence value to be $1.9 \times 10^{19} n/cm^2$ (E > 1.0 MeV). This value is acceptable because the calculated dosimetry reaction rates and the corresponding measured values are close, thus lending credibility to the projected fluence value.

The vessel material properties were estimated based on this fluence value. The methodology used for the estimation of the P-T curves and the LTOP setpoints complies with the method described in WCAP-14040 which has been approved by the NRC (Ref. 5). The P-T limits were based on the ASME Appendix G, Section XI Code Case N-640.

The method used provides for staggered PORV setpoints to avoid excessive transient undershoot and addresses single failure. Instrument uncertainty is included in the estimation of the high setpoint and the pressure undershoot. Both mass and energy addition transients were analyzed.

Because the method has been approved by the staff and the licensee has applied the method correctly by considering all of the major physical phenomena, the staff finds the proposed P-T curves and LTOP setpoints acceptable.

In summary, the licensee's proposed P-T Limits and Overpressure Protection System TS changes resulted from the reevaluation of projected fluence values because of the removal of surveillance capsule X and the application and use of ASME Code Case N-640 to satisfy the Appendix G requirements. The resulting TS, associated action statements, and surveillance requirements reflect the revised analyses. The staff finds that the value of the fluence used in the estimation of the material properties is acceptable. The revised TS LCOs provide equivalent protection and conform with the STSs. Therefore, the staff finds the proposed TS changes to be 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 State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* on July 11, 2001 (66 FR 36340). Accordingly, based on the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant impact on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, 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 activities 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 to the health and safety of the public.

7.0 REFERENCES

1. WCAP-15405, "Analysis of Capsule X from the Northeast Nuclear Energy Company Millstone Unit 3 Reactor Vessel Radiation Surveillance Program" by E. Terek, et al., Westinghouse Electric Corporation, LLC, May 2000.

2. DOORS 3.1, "RSICC Computer Code Collection CCC-650, "DOORS 3.1, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

3. BUGLE-96, RSICC Data Library Collection DLC-185 “BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” March 1996.

4. RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” U.S. Nuclear Regulatory Commission, March 2001.

5. WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” dated January 1996.

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