UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 29, 2015

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISE OR ADD ACTIONS TO PRECLUDE ENTRY INTO LIMITING CONDITION FOR OPERATION 3.0.3 CONSISTENT WITH TSTF-426, REVISION 5 (TAC NO. MF4471)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No. 321 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated June 30, 2014, as supplemented by letter dated January 29, 2015.

The amendment revises the Technical Specifications (TSs) by adopting approved Technical Specification Task Force (TSTF) traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 – RITSTF Initiatives 6b and 6c," and providing a short completion time to restore an inoperable system for conditions under which the previous TS required a plant shutdown. The amendment also reformats each proposed MPS2 custom TS ACTION format to a two-column tabular format.

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,

Richard V. Guzman, Sr. Project Manager
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 321 to DPR-65
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 321
Renewed License No. DPR-65

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

   A. The application for amendment by the applicant dated June 30, 2014, as supplemented by letter dated January 29, 2015, 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 Renewed Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 321, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael I. Dudek, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: July 29, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 321

RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

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
3/4 4-3	3/4 4-3
---	3/4 4-3a
---	3/4 4-3b
3/4 4-4	3/4 4-4
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13
3/4 6-25	3/4 6-25
3/4 7-16	3/4 7-16
---	3/4 7-16a
---	3/4 7-16b

Connecticut, in accordance with the procedures and limitations set forth in this renewed operating license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2700 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 321, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

LIMITING CONDITION FOR OPERATION

3.4.3 Both power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

Inoperable Equipment	Required ACTION
a. One or both PORVs, capable of being manually cycled.	a.1 Within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s)*; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
b. One PORV, not capable of being manually cycled.	b.1 Within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
c. - - - - NOTE - - - - Not applicable when a second PORV intentionally made inoperable. - - - - - - - - - - - Two PORVs, not capable of being manual cycled.	c.1 Close the associated block valves within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND c.2 Remove power from associated block valves within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND

* The block valve(s) may be stroked, as necessary, during plant cooldown to prevent thermal binding.

ACTION: (continued)

Inoperable Equipment	Required ACTION
c. (continued)	c.3 Verify LCO 3.7.1.2, "Auxiliary Feedwater Pumps," is met within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND c.4 Restore at least one PORV to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
d. One block valve.	d.1 Prevent its associated PORV from opening automatically within 1 hour, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND d.2 Restore the block valve to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
e. - - - - NOTE - - - - Not applicable when second block valve intentionally made inoperable. - - - - - - - - - - - Two block valves.	e.1 Verify LCO 3.7.1.2, "Auxiliary Feedwater Pumps," is met within 1 hour; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND e.2 Restore at least one block valve to OPERABLE status within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. Once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. Once per 18 months by performance of a CHANNEL CALIBRATION.

c. Once per 18 months by operating the PORV through one complete cycle of full travel at conditions representative of MODES 3 or 4.

4.4.3.2 Each block valve shall be demonstrated OPERABLE once per 92 days by operating the valve through one complete cycle of full travel. This demonstration is not required if a PORV block valve is closed in accordance with the ACTIONS of Specification 3.4.3.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 70%, and

b. At least two groups of pressurizer heaters each having a capacity of at least 130 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Inoperable Equipment	Required ACTION
a. Pressurizer water level not within limit.	a.1 Be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.
b. One group of pressurizer heaters.	b.1 Restore the inoperable group of pressurizer heaters to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.
c. - - - - NOTE - - - - Not applicable when second group of required pressurizer heaters intentionally made inoperable. - - - - - - - - - - - Two groups of pressurizer heaters.	c.1 Restore at least one group of pressurizer heaters to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water level shall be determined to be within its limits at least once per 12 hours.

4.4.4.2 Verify at least two groups of pressurizer heaters each have a capacity of at least 130 kW at least once per 92 days.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains, with each cooling train consisting of two containment air recirculation and cooling units, shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

Inoperable Equipment	Required ACTION
a. One containment spray train	a.1 Restore the inoperable containment spray train to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia within the following 6 hours.
b. One containment cooling train	b.1 Restore the inoperable containment cooling train to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.
c. One containment spray train AND One containment cooling train	c.1 Restore the inoperable containment spray train or the inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
d. - - - - NOTE - - - - Not applicable when second containment spray train intentionally made inoperable. - - - - - - - - - - - Two containment spray trains.	d.1 Verify LCO 3.7.6.1, "Control Room Emergency Ventilation System," is met within 1 hour or be in HOT SHUTDOWN within the next 12 hours. d.2 Restore at least one inoperable containment spray train to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 12 hours.
e. Two containment cooling trains	e.1 Restore at least one inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
f. All other combinations	f.1 Enter LCO 3.0.3 immediately.

* The Containment Spray System is not required to be OPERABLE in MODE 3 if pressurizer pressure is < 1750 psia.

SURVEILLANCE REQUIREMENTS

4.6.2.1.1 Each containment spray train shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying each containment spray manual, power operated, and automatic valve in the spray train flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. By verifying the developed head of each containment spray pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

c. At least once per 18 months by verifying each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

d. At least once per 18 months by verifying each containment spray pump starts automatically on an actual or simulated actuation signal.

e. By verifying each spray nozzle is unobstructed following activities that could cause nozzle blockage.

4.6.2.1.2 Each containment air recirculation and cooling unit shall be demonstrated OPERABLE:

a. At least once per 31 days by operating each containment air recirculation and cooling unit in slow speed for $\geq$ 15 minutes.

b. At least once per 31 days by verifying each containment air recirculation and cooling unit cooling water flow rate is $\geq$ 500 gpm.

c. At least once per 18 months by verifying each containment air recirculation and cooling unit starts automatically on an actual or simulated actuation signal.

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent Enclosure Building Filtration Trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Inoperable Equipment	Required ACTION
a. One Enclosure Building Filtration Train.	a.1 Restore the inoperable Enclosure Building Filtration Train to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.
b. - - - - NOTE - - - - Not applicable when second Enclosure Building Filtration Train intentionally made inoperable. - - - - - - - - - - - Two Enclosure Building Filtration Trains.	b.1 Verify at least one train of containment spray is OPERABLE within 1 hour or be in COLD SHUTDOWN within the next 36 hours. AND b.2 Restore at least one Enclosure Building Filtration Train to OPERABLE status within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each Enclosure Building Filtration Train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal absorber train and verifying that the train operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal absorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent Control Room Emergency Ventilation Trains shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6.

During movement of recently irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

Inoperable Equipment	Required ACTION
a. One Control Room Emergency Ventilation Train, except as specified in ACTION c.	a.1 Restore the inoperable Control Room Emergency Ventilation Train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
b. - - - - NOTE - - - - Not applicable when second Control Room Emergency Ventilation Train intentionally made inoperable. - - - - - - - - - - - Two Control Room Emergency Ventilation Trains, except as specified in ACTION c.	b.1 Initiate action to implement mitigating actions immediately or be in HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours. AND b.2 Verify LCO 3.4.8, "Reactor Coolant System, Specific Activity," is met within 1 hour or be in HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours. AND b.3 Restore at least one Control Room Emergency Ventilation Train to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

* The Control Room Envelope (CRE) boundary may be opened intermittently under administrative control.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

MODES 1, 2, 3, and 4:

Inoperable Equipment	Required ACTION
c. One or more Control Room Emergency Ventilation Trains, due to an inoperable CRE boundary.	c.1 Immediately initiate action to implement mitigating actions or be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. AND c.2 Verify, within 24 hours, mitigating actions ensure CRE occupant exposures to radiological and chemical hazards will not exceed limits, and mitigating actions are taken for exposure to smoke hazards or be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. AND c.3 Restore CRE boundary to OPERABLE status within 90 days or be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

MODES 5 and 6, and during movement of recently irradiated fuel assemblies:**

Inoperable Equipment	Required ACTION
d. One Control Room Emergency Ventilation Train, except due to an inoperable CRE boundary.	d.1 Restore the inoperable Control Room Emergency Ventilation Train to OPERABLE status within 7 days. AND d.2 After 7 days, initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation Train in the recirculation mode of operation or immediately suspend the movement of recently irradiated fuel assemblies.
e.1 Both Control Room Emergency Ventilation Trains, OR e.2 The OPERABLE Control Room Emergency Ventilation Train required to be in the recirculation mode by ACTION d. not capable of being powered by an OPERABLE normal and emergency power source, OR e.3 One or more Control Room Emergency Ventilation Trains, due to an inoperable CRE boundary.	e.1 Immediately suspend the movement of recently irradiated fuel assemblies.

** In MODES 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then ACTION 3.7.6.1.d or 3.7.6.1.e shall be invoked as applicable.

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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 321

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated June 30, 2014, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14188B189), supplemented by letter dated January 29, 2015, (ADAMS Accession No. ML15035A060), Dominion Nuclear Connecticut, Inc. (the licensee), submitted a License Amendment Request for Millstone Power Station Unit No. 2 (MPS2). The amendment revises the Technical Specifications (TSs) by adopting approved Technical Specification Task Force (TSTF) traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into [Limiting Condition for Operation] LCO 3.0.3 – RITSTF Initiatives 6b and 6c" (Reference 3), and providing a short completion time to restore an inoperable system for conditions under which the previous TS required a plant shutdown. The amendment also reformats each proposed MPS2 custom TS ACTION format to a two-column tabular format. Revision 5 of TSTF-426 was issued in the *Federal Register* (FR) on May 30, 2013 (78 FR 32476).

The supplemental letter dated January 29, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the FR on December 23, 2014 (79 FR 77044).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance which the NRC staff considered in assessing the proposed TS change are as follows:

The Commission's regulatory requirements related to the content of the TS are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." Pursuant to 10 CFR 50.36(c) the TS are required to include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation at 10 CFR 50.36(c)(2)(i) states: "When [an LCO] of a nuclear reactor is not met,

the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

The staff also used NUREG-1432, Revision 4, "Standard Technical Specifications [STS], Combustion Engineering Plants," (Reference 4) as guidance.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" Rev. 2 (Reference 5), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. RG 1.174 also provides risk acceptance guidelines for evaluating the results of such evaluations.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 6). Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed change meets the following key principles:

1. The proposed change meets the current regulations, unless it is explicitly related to a requested exemption.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency (CDF) or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (60 FR 42622).

5. The impact of the proposed change should be monitored using performance measurement strategies.

The NRC staff reviewed the licensee's proposed changes against (1) the requirements of 10 CFR 50.36; (2) the corresponding changes made to the STS by TSTF-426, Revision 5, which were approved for adoption in the Notice of Availability of TSTF-426 issued in the FR on May 30, 2013 (78 FR 32476); and (3) the methodology approved in Topical Report (TR) WCAP-16125 (Reference 7), as documented in a safety evaluation dated May 24, 2010 (Reference 8). The TR WCAP-16125 was reviewed against RG 1.174 and SRP Section 19.2.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

In addition to the TR WCAP-16125 justified modifications discussed below the ACTION statements for the affected TS have been formatted into a two-column table consisting of a column describing the "Inoperable Equipment" and a column describing the "Required

ACTION." The licensee asserts that "the two-column table organizes the conditions and associated actions in a manner that eliminates ambiguity" and that the tabular format achieves "a transition to the two column format in a manner consistent with NUREG-1432."

TR WCAP-16125 justified modifications to various TS to add a Condition for loss of redundant features representing a loss of safety function for a system or component included within the scope of the plant TS. It would replace Required ACTIONs (RA) requiring either a default shutdown or explicit LCO 3.0.3 entry with a RA based on the risk significance for the system's degraded condition. The Condition being added is for redundant trains discovered to be inoperable. The Condition only applies to discovery of an emergent condition resulting in redundant trains being inoperable, not from the second train intentionally being made inoperable. The completion times (CT) associated with the proposed actions are specified. The CTs are intentionally of short duration to allow for restoring the system to an operable condition, thereby avoiding the risk associated with an immediate controlled shutdown. In all but the TS change for reactor coolant system pressurizer power operated relief valves (PORVs), a 24-hour CT is justified. The CT for the pressurizer PORV specification is 8 hours. Table 1 summarizes the proposed TS changes.

Table 1

TS	SYSTEM/COMPONENT	CONDITION	CURRENT CT	PROPOSED CT
3/4.4.4	Reactor Coolant System, Pressurizer	Two groups of Class 1E heaters inoperable	None/ LCO 3.0.3 *default shutdown*	24 hours
3/4.3.4.3	Reactor Coolant System, Relief Valves	Two PORVs inoperable and not capable of being cycled manually; or Two block valves inoperable	1 hour	8 hours (1)
3/4.6.2.1	Containment Systems Depressurization and Cooling Systems, Containment Spray and Cooling Systems (CSCS)	Two containment spray trains inoperable	None, LCO 3.0.3	24 hours (2)
3/4.6.5.1	Containment Systems Secondary Containment, Enclosure Building Filtration System (EBFS)	Two trains inoperable	None, LCO 3.0.3	24 hours (3)
3/4.7.6.1	Plant Systems, Control Room Emergency Ventilation System (CREVS)	Two trains inoperable (Modes 1-4) for reasons other than an inoperable boundary	1 hour	24 hours (4)

(1) Must include verification that the LCO for auxiliary feedwater is met, which requires both trains to be operable. In addition, the new 8-hour CT does not apply in the STS to PORVs which are leaking and unisolable.
(2) Must include verification that the LCO for CREVS is met.
(3) Must include verification that at least one train of the CS system is operable.
(4) Must include verification that LCO 3.4.8, "Reactor Coolant System Specific Activity," is met.

The licensee reviewed the NRC staff's model safety evaluation dated May 20, 2013 (ADAMS Accession No. ML13036A381). Based on its review, the licensee concluded that the justifications presented in the TSTF-426 proposal and the staff's model safety evaluation for TSTF-426, are applicable to MPS2 and can be applied to justify the incorporation of the proposed changes to the MPS2 TSs.

3.2 Conformance with the Five Key Principles of SRP Section 19.2 as Summarized in the Safety Evaluation of TR WCAP-16125

The changes proposed in TSTF-426 are consistent with Commission-approved TR WCAP-16125. In Reference 5, the NRC staff evaluated TR WCAP-16125 for conformance with the five key principles of SRP Section 19.2.

3.2.1 Compliance with Current Regulations

The regulations at 10 CFR 50.36 require either a plant shutdown or other remedial actions specified by TSs when an LCO is not met. The proposed change provides new action requirements for conditions of equipment inoperability which currently require an immediate plant shutdown. The NRC staff finds that since such remedial actions are permitted per 10 CFR 50.36, the proposed change continues to comply with current regulations, and therefore, satisfy this key principle.

3.2.2 Defense-in-Depth

The NRC staff finds that the proposed change addresses conditions where both trains of a system are inoperable, resulting in a loss of that system's function and a temporary reduction in the defense-in-depth capabilities of the plant. Each proposed change addresses the remaining available alternative system(s) capable of providing mitigation of events, and where applicable, includes requirements to assure these required backup systems are operable. The reduced level of defense-in-depth is retained by verification that both trains (if applicable) of the backup system are operable. Therefore, the NRC staff finds that this key principle is satisfied by the unique requirements identified for each proposed TS change.

3.2.3 Safety Margins

The NRC staff finds that the proposed change does not have any impact on the use of NRC-approved codes and standards, nor do the proposed changes impact any acceptance criteria used in a plant's licensing basis. Under the current TS, if an accident occurs during the 6-hour controlled shutdown time of LCO 3.0.3 caused by two trains of these systems being unavailable, it could potentially result in offsite dose limits that do not meet NRC regulatory limits. Since the proposed changes do not modify the design basis of the systems evaluated, extending the allowed outage time to 24 hours would have no quantitative effect on the dose consequence as compared to the existing condition. As such, the NRC staff finds that the proposed changes would not significantly reduce the plant's available safety margin, and therefore, this key principle is satisfied.

3.2.4 Performance Monitoring

The NRC staff finds that the proposed change would permit continued plant operation for short periods to address emergent equipment failures. Degradation of equipment performance could lead to excessive use of the new action requirements. The NRC staff finds that this is adequately addressed by equipment performance monitoring required by 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and therefore, this key principle is satisfied.

3.2.5 Risk Assessment

The risk of each of the TS LCOs for which action requirements are proposed is evaluated in TR WCAP-16125 by three methods, as described below.

Method 1:

For those TSs governing systems or components which provide mitigation of core damage and large early releases, changes in the core damage frequency, and large early release frequency (ΔCDF and ΔLERF, respectively) metrics are calculated using a simplified generic method, and the results are compared to the acceptance guidelines of RG 1.174. This applies to TS LCO 3.4.3, Reactor Coolant System, Relief Valves.

For calculations of ΔCDF, a bounding approach was applied to evaluate loss of function of a system by identifying the initiating events for which the system provides mitigation, and assuming that the event goes directly to core damage. No credit was taken for alternate mitigation strategies, and the baseline CDF was effectively assumed to be zero. The initiating event frequencies were taken from NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995" (Reference 9). The licensee verified that initiating frequencies in NUREG/CR-5750 are bounding for MPS2.

For ΔLERF, a simplified approach using an event tree was developed to calculate the fraction of core damage events which result in large early releases. The event tree assessed containment isolation status, reactor coolant system (RCS) pressure, secondary side depressurization via the steam generators, thermally-induced steam generator tube rupture (SGTR), and reactor pressure vessel (RPV) lower head failure. Assumptions related to the potential impact on LERF for each of these events, and the associated basis for probabilities used in the analysis, are discussed below:

Containment Isolated – This event defines containment integrity prior to the core damage event. If containment is not isolated, then a large early release will result concurrent with core damage. A probability of 3.0E-3 was applied for an unisolated containment, which is identified as the upper end of the range used in the [Combustion Engineering] CE Probabilistic Risk Assessment (PRA) models in TR WCAP-16125.

RCS Pressure – High – This event defines the RCS pressure at the time of core damage. If the pressure is low, then large early releases are assumed not to occur (except via an unisolated containment); otherwise, thermally-induced SGTR and high pressure melt ejection events are further evaluated. All core damage events involving

loss-of-coolant accidents (LOCAs) are assumed to result in low or intermediate RCS pressure, and all other events result in high RCS pressure.

Steam Generator Depressurization – This event defines the status of the secondary side, and affects the next event which is the potential for induced SGTR. Depressurization of the secondary side occurs either due to prior operator response or due to failure of a safety relief valve. Based on NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture" (Reference 10), a probability of 0.9 is assigned for secondary depressurization.

Thermally-Induced SGTR Occurs – This event represents a loss of steam generator tube integrity due to thermal stresses during a severe accident, which is assumed to result in a large early release. Two values are used, based on the status of the prior event, for steam generator depressurization. A probability of 0.5 is assigned when the steam generators are depressurized, and 0.01 otherwise. These values are conservative, based on the assumptions regarding tube age and integrity and based on neglecting operator actions to depressurize the RCS after core damage.

RPV Lower Head Failure Results in Containment Failure – This event represents a high pressure failure of the lower head, with an energetic discharge of the molten fuel and direct containment heating, leading to failure of containment. Based on NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments" (Reference 11), the conditional containment failure probability given the event for CE-designed plants is 0.01, which is considered to be a bounding value.

None of the assessed initiating events include either SGTRs or other containment bypass events because the systems being evaluated do not mitigate these events. The NRC staff concludes that the simplified LERF event tree is reasonable and acceptable to support the evaluation of LERF for the scope of TR WCAP-16125.

Method 2:

For TS LCO 3.4.4, Reactor Coolant System, Pressurizer Heaters, an evaluation of the increased likelihood of a plant trip due to degraded pressure control is made in order to calculate ΔCDF. The ΔLERF calculation for this TS is the same simplified approach described above for Method 1.

Method 3:

The remaining systems (and associated TS) associated with mitigation of radiological releases with magnitudes less than those associated with LERF are: CSCS (TS 3.6.2.1), EBFS (TS 3.6.5.1), and CREVS (TS 3.7.6.1). There is no impact to either CDF or LERF, as the systems are provided to meet design basis dose limits. EBSF is functionally similar to the Shield Building Exhaust Air Cleanup System (SBEACS); and CREVS is functionally similar to the Control Room Emergency Air Cleanup System (CREACS). SBEACS and CREACS were evaluated in TR WCAP-16125 and the TSTF-426 model safety evaluation. As

described in TR WCAP-16125, an evaluation of the frequency of events which challenge the systems was made and compared to the acceptance guidelines of RG 1.174 applicable to ΔLERF in order to characterize the risk of these lesser releases. In TR WCAP-16125 additional justification is provided based on the availability of other systems which provide a degree of defense-in-depth for prevention of these releases.

To assess the impact of the unavailability of these systems, TR WCAP-16125 examined the expected iodine releases for three categories of events:

- Beyond design basis scenarios that lead to large early releases,
- Maximum Hypothetical Accident (MHA), and
- LOCA and Non-LOCA Design-Basis Accidents (DBA).

The purpose of this assessment was to show that, using worst case assumptions, the potential accident releases anticipated under the short-term operational conditions proposed by the increased CT for the CSCS and EBFS will be well below and bounded by a large early release. For clarity, the TR WCAP-16125 evaluation was limited to the release of iodine. For each category, iodine releases were estimated assuming various combinations of system availability. The results of this assessment are shown in Table 4.3-1 of TR WCAP-16125, supplemented by RAI responses (References 12 & 13). The NRC staff reviewed the assumptions and methodology used to determine the bounding iodine release quantities and resulting dose consequences and found that in all cases appropriately conservative assumptions were used.

To reduce the impact of an increased CT for the CREVS, TR WCAP-16125 added conditions to verify that RCS specific activity is within limits and to verify that dose mitigating actions are available in the CR. For limited durations, such as the short-term operational conditions proposed by the increased CT for the CREVS, the NRC staff has accepted credit for the use of respirators and potassium iodide on an interim basis to demonstrate that control room dose limits can be met. The 24-hour CT proposed in TR WCAP-16125 for the CREVS is consistent with the allowed 24-hour period for the evaluation of a breach of the control room envelope provided in Traveler TSTF-448 (Reference 13).

Based on an evaluation of methods and assumptions used to produce the results shown in Table 4.3-1 of TR WCAP-16125, the NRC staff has reasonable assurance that the postulated accident releases calculated for the short-term operational conditions proposed by the increased CT for the CSCS and EBFS will be well below the LERF releases. In addition, the NRC staff has reviewed the bases for the increased CT for the CREVS and has determined that the proposed conditions and compensatory measures provide reasonable assurance that control room habitability will be adequately maintained during the proposed 24-hour CT.

External events, internal fires, and floods, were not evaluated in TR WCAP-16125. None of the systems being evaluated provide a primary mitigating function for external events, and therefore these events are not significant to the risk-informed decision.

The TR WCAP-16125 also evaluated sensitivity studies for key areas of uncertainty in the analyses. Specifically, TR WCAP-16125 considered uncertainties in the initiating event frequencies (which are the input to the CDF calculations) and showed that even when assuming a 95 percent upper bound frequency, the uncertainties would not result in excessive risk. These were also propagated into the LERF calculations with similar results. The TR WCAP-16125 also addressed uncertainties in the thermally-induced SGTR assumptions and steam generator depressurization assumptions, and demonstrated that the LERF results are not significantly impacted. These sensitivity studies which were performed to evaluate the key sources of uncertainty in the risk analyses, adequately demonstrate the robustness of the results to support the proposed TS changes.

3.3 NRC Staff Evaluation of Proposed TS Changes

In addition to the detailed evaluation of each proposed TS change below, the ACTION statements for the affected TS have also been formatted into a two-column table consisting of a column describing the "Inoperable Equipment" (IE) and a column describing the "Required ACTION" (RA). As stated in their application dated June 30, 2014, the licensee asserts that "the two-column table organizes the conditions and associated actions in a manner that eliminates ambiguity" and that the tabular format achieves "a transition to the two column format in a manner consistent with NUREG-1432."

3.3.1 TS 3.4.4

The pressurizer and the Class 1E electrical heaters maintain a liquid-to-vapor interface to permit RCS pressure control during normal operations and in response to anticipated design basis transients. The Class 1E heaters, with their power provided by emergency AC power busses, are used to maintain RCS subcooling during a natural circulation cooldown, and the unavailability of the heaters will extend the time to reach entry conditions for the shutdown cooling system. The unavailability of the Class 1E heaters may complicate steady-state RCS pressure control and may increase the potential of an unplanned reactor trip. However, the availability of additional heaters beyond the two groups required by this TS LCO permit continued RCS pressure control.

The current LCO 3.4.4 does not provide any action requirements for two groups of pressurizer heaters inoperable; therefore, ACTION b. applies, which requires an immediate plant shutdown.

The licensee proposes to modify the ACTION statements into a two column tabular format consisting of a column describing the "Inoperable Equipment," and another column describing the "Required ACTION":

- Original ACTION a. becomes Required ACTION (RA) b.1, for Inoperable Equipment (IE) "One group of pressurizer heaters."

- Original ACTION b. is slightly modified and becomes RA a.1 addressing the IE condition of "Pressurizer water level not within the limit."

- RA c.1 is newly added to address the IE condition of "Two groups of pressurizer heaters" using a 24-hour CT.

The IE designator used in this two column format is similar to the "CONDITION" designator used in NUREG-1432, "Standard Technical Specifications," while the two column RA designator encompasses the functions of both the "REQUIRED ACTION" and "COMPLETION TIME" designators used in NUREG-1432.

The proposed change allows continued operation under an existing action requirement for two groups of pressurizer heaters that are inoperable. The unavailability of the Class 1E pressurizer heaters would not have any significant impact on plant transient response, and so there is no quantifiable impact to CDF or LERF. While mitigation of a SGTR is enhanced by the availability of pressurizer heaters, the non-Class 1E heaters can also function if offsite power is available, and plant procedures provide for mitigation of a SGTR without pressurizer heaters, if necessary.

Conservatively, the risk result due to increased likelihood of a reactor trip was calculated by assuming an order-of-magnitude increase in the reactor trip frequency when both Class 1E heaters are inoperable. The risk result is then calculated based on the conditional core damage probability given a reactor trip with no other complications:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.0E-7/yr	<1.0E-6/yr	3.8E-9/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on a bounding once per three year entry into the proposed action requirement from TR WCAP-16125 and assumed that the entire 24-hour duration of the CT is used. The risk results are well below the acceptance guidelines of RG 1.174 as noted in the table.

Minimum pressurizer heater capability is supplemented by the normal availability of non-Class 1E heaters for normal plant pressure control, and the availability of plant procedures which provide plant shutdown and cooldown guidance with or without pressurizer heaters. If the available heaters are sufficient to maintain RCS pressure control, normal plant operations can continue. Because unavailability of Class 1E and non-Class 1E heaters would physically result in plant shutdown, the NRC staff does not consider it necessary to specify additional TS or administrative requirements for the non-Class 1E heater availability.

The current TS LCO 3.4.4 does not contain an ACTION for two required groups of pressurizer heaters inoperable. As a result ACTION b. applies which requires an immediate plant shutdown. The new IE condition c. is being added for two required groups of pressurizer heaters inoperable. The associated RA requires restoration of one group of heaters to operable status within 24 hours. The new IE condition c. is modified by a note stating it is not applicable when the second group of required pressurizer heaters is intentionally made inoperable. The new IE condition a. applies for pressurizer water level not within the limits with the same shutdown required action as was specified in existing ACTION b. Any other pressurizer

inoperable condition would now result in entry into LCO 3.0.3. Therefore, LCO 3.0.3 still results in a shutdown allowing one additional hour for shutdown preparations and, therefore, is acceptable.

The current ACTION a. is relocated into the tabular format as IE b. and RA b.1 with the requirements essentially unchanged and is acceptable.

The conservatively-calculated risk result is within the acceptance guidelines of RG 1.174, and there is limited impact of plant shutdown and cooldown without pressurizer heaters. Therefore, the NRC staff finds the proposed TS changes, proposed new action requirement, and proposed 24-hour CT acceptable.

3.3.2 TS 3.4.3

The PORVs and associated block valves are required to be operable to minimize the potential for a small-break LOCA through a PORV pathway. The PORVs automatically open for RCS pressure control to avoid challenging the primary safety relief valves, and may be manually opened by the operator to control pressure. In the event of a total loss-of-feedwater to the steam generators (SGs), one or more PORVs may be opened manually to provide for feed-and-bleed cooling of the reactor using once-through cooling from high pressure injection to the RCS. The PORVs may also be used for low temperature overpressure protection during heatup and cooldown. The PORV may be manually operated to depressurize the RCS in response to normal or abnormal transients. The PORV may be used for depressurization when pressurizer spray is not available, a condition that may be encountered during a loss-of-offsite power (LOOP). The PORVs can be manually operated to reduce RCS pressure in the event of a SGTR with a LOOP.

When two PORVs are inoperable and not capable of being manually cycled the change provides an 8-hour CT to restore at least one PORV or one block valve to operable status before shutdown. This action may only be applied provided the PORV is isolable by its block valve.

The current TS requirement with both PORVs inoperable and incapable of being manually cycled is to restore at least one PORV to operable status within 1 hour or close the associated block valve and remove power from that block valve. A default plant shutdown is required. The risk result of unavailable PORVs or block valves is primarily attributable to the non-design basis function of providing for feed-and-bleed cooling:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.5E-7/yr	<1.0E-6/yr	1.1E-8/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on a bounding once per three-year entry into the proposed action requirement from TR WCAP-16125 and assumed that the entire 8-hour duration of the CT is used. The risk results are well below the acceptance guidelines of RG 1.174 as noted in the table.

The primary safety relief valves provide the design basis pressure control function, controlled by TS requirements. The non-design basis feed-and-bleed function is considered to be risk significant, and the proposed change includes a TS requirement to confirm that the LCO for auxiliary feedwater (AFW) is met, which requires both trains to be operable. A reduced level of defense-in-depth is retained by verification of the operability of the both AFW pumps. These requirements assure that mitigation capability is available for those design-basis accidents (DBAs) or anticipated operational occurrences requiring the pressure control and heat removal functions of the PORVs.

In addition, the new 8-hour CT does not apply to PORVs which are leaking and unisolable.

The following proposed changes to the TS are being made:

- Current TS LCO 3.4.3, ACTIONs a. and b. transfer to the two column format essentially unchanged as new IE conditions a. and b. with corresponding RAs a.1 and b.1.

- Current ACTION c. for two PORVs inoperable and not capable of being manually cycled is transferred to the 2 column format as IE condition c. "Two PORVs not capable of being manually cycled" in the IE column. Corresponding RAs c.1, c.2, c.3, and c.4 in the RA column are actions that must be taken. The RAs c.1 and c.2 are transferred from the current format to the two column format essentially unchanged. The RA c.3 is the new defense-in-depth addition to verify LCO 3.7.1.2, "Auxiliary Feedwater," is met within 1 hour. The RA c.4 is transferred to the two-column format and additionally revised to use the new 8-hour COMPLETION TIME to restore at least 1 PORV to operable status. The new IE condition c. is modified by a note stating it is not applicable when a second PORV is intentionally made inoperable.

- Existing ACTION d. actually addresses two IE conditions: (1) one block valve inoperable; and (2) two block valves inoperable. These conditions are split into IE entries d. and e. so that the RAs can be specified separately. The IE condition d. for one block valve inoperable specifies RAs d.1 and d.2 which maintain the same completion time as the original ACTION d. The IE e. for two block valves inoperable specifies RAs e.1 and e.2. In RA e.1 there is a new defense-in-depth addition to verify LCO 3.7.1.2, "Auxiliary Feedwater," is met within 1 hour. The RA e.2 implements the new 8 hour time to restore one block valve to operable status. The new IE condition e. is additionally modified by a note stating it is not applicable when a second block valve is intentionally made inoperable.

The licensee confirmed that the pressurizer PORV TS contains requirements equivalent to NUREG-1432 with regard to leaking and unisolable PORVs. The NRC staff has determined that the TS are equivalent and, therefore, acceptable.

3.3.3 TS 3.6.2.1

The containment spray (CS) system and the containment coolers provide containment heat removal following accidents which release high energy steam to the containment. In addition to the heat removal function, the CS system enhances post-accident fission product removal. Each train of the CS system provides a nominal 50 percent of the cooling function, and similarly

each train of the containment coolers provides 50 percent of the cooling function; thus the combined capacity of both systems is 200 percent.

An explicit LCO 3.0.3 entry is provided in TS LCO 3.6.2.1 when less than 100 percent containment cooling capacity is available (i.e., any combination of three or more trains inoperable).

For TS LCO 3.6.2.1, when both CS trains are inoperable, and therefore, the fission product removal function is not available, an explicit LCO 3.0.3 entry is required. The RAI responses to the TR WCAP-16125 (References 12 & 13) proposed a 24-hour CT for TS LCO 3.6.2.1 consistent with the other iodine removal TS changes. The RAI responses also identified that the TS-required operability of the containment coolers would provide a similar iodine removal function such that additional TS requirements for operability of other iodine removal systems would not be required. A TS action for operability of the CREVS was proposed to assure additional defense-in-depth for control room functionality when both CS trains are inoperable during the 24-hour CT.

Based on the information in TR WCAP-16125, it may be conservatively assumed that if both CS trains are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event which would require the unavailable system would be:

$$(1E\text{-}4/\text{year}) \times (24 \text{ hours}) \times (\text{year}/8760 \text{ hours}) = 2.7E\text{-}7$$

Assuming a once per three-year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore, provides a context for consideration of the risk result for smaller releases.

When the function of the CS system for fission product removal is unavailable, then the operability of the CREVS (which provides for filtration to protect control room habitability) will be verified as a defense-in-depth measure.

The TS LCO 3.6.2.1 is already in the two column format due to a previously approved change. It contains a generic existing RA for IE condition e. (which applies when either two containment spray trains are inoperable or any combination of three or more trains are inoperable). It currently requires entering LCO 3.0.3 immediately for two inoperable CS trains. The proposed change adds a new IE condition d. for "Two containment spray trains" inoperable. This contains two RAs d.1 and d.2 both of which must be completed to apply this IE condition. The RA d.1 is to verify within 1 hour that LCO 3.7.6.1 for two independent CREVS trains be met and RA d.2 to restore one CS train to operable status within 24 hours. The IE condition d. is additionally modified by a note stating it is not applicable when the second containment spray train is intentionally made inoperable. Existing IE condition d. is then renamed IE condition e. and existing IE condition e. is renamed IE condition f.

The TR WCAP-16125 states that IE condition d. is applicable when two containment spray trains are inoperable provided that at least one containment air cooler is operable. This restriction is imposed by IE condition f., which addresses any other combination with a RA (f.1) to enter LCO 3.0.3 immediately. Existing RAs d.1 and e.1 are renamed RAs e.1 and f.1 respectively.

The zero risk result for severe accidents is well below the acceptance guidelines of RG 1.174, and there is verification of operability of the CREVS. Therefore, the NRC staff finds a new action requirement with a 24-hour CT would be acceptable for the case of both CS trains inoperable.

3.3.4 TS 3.6.5.1

The EBFS functions to assure radioactive material released from containment leakage following a DBA is filtered prior to being exhausted to the environment. This system includes two redundant trains with high efficiency particulate air filters, moisture absorbers, and charcoal adsorbers in the flowpath. The EBFS filters leakage from containment into the enclosure building (functionally similar to the shield building). The design basis for this system is a postulated MHA involving a LOCA with a short duration uncovery of fuel, resulting from a temporary interruption, or significant degradation, of the ECCS flow. The event is assumed to result in significant iodine releases (40 – 50 percent of core inventory) from the fuel into the containment. The containment remains intact, with no more than the design basis leakage permitted by TSs. Releases associated with the MHA are significantly below the release which would occur for a postulated large early release (at least two orders of magnitude lower). This system does not provide any mitigation capability for preventing either core damage or large early releases.

The current TSs do not address the condition of two inoperable trains of this system; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour time to restore at least one train of the affected system to operable status in order to permit continued operation under an existing action requirement.

As noted above, this system does not provide any core damage or large early release mitigation. Therefore, the risk results are zero for these systems. However, it may be conservatively assumed that if this system is unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event which would require the unavailable system, would be:

$$(1E\text{-}4/\text{year}) \times (24 \text{ hours}) \times (\text{year}/8760 \text{ hours}) = 2.7E\text{-}7$$

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore provides a context for consideration of the risk result for smaller releases.

As noted in TR WCAP-16125, there are also higher frequency DBAs (e.g., rod ejection and reactor coolant pump locked rotor) which are assumed to result in fuel damage and therefore rely upon this system to filter any containment leakage. These accidents are associated with releases from the fuel into containment two or more orders of magnitude below those associated with the MHA described above, and four or more orders of magnitude below large early releases.

Containment spray can effectively scrub the post-accident containment atmosphere of fission products, and therefore reduce reliance upon the downstream air cleanup systems. In order to assure additional defense-in-depth protection for the spectrum of accidents (for which this system provides mitigation), the TS action will include a verification of operability of at least one train of the CS system.

Currently TS 3.6.5.1 does not contain an ACTION for both EBFS trains inoperable. As a result, this condition would require immediate entry into LCO 3.0.3. The following proposed TS changes are being made:

- The existing ACTION for one EBFS train inoperable is transferred into the two column format essentially unchanged as IE condition a. with associated Required Action a.1.

- A new IE condition b. is added which applies when two EBFS trains are inoperable with associated RAs b.1 and b.2. The RA b.1 requires verification within 1 hour that at least one train of containment spray is operable. The RA b.2 allows 24 hours to restore at least one EBFS train to operable status. Additionally, IE condition b. is modified by a note stating it is not applicable when the second EBFS train is intentionally made inoperable.

The zero risk result for severe accidents is well below the acceptance guidelines of RG 1.174, and there is an additional restriction on operability of at least one CS train in the TS. Therefore, the NRC staff finds the proposed new action requirement and the 24-hour CT are acceptable.

3.3.5 TS 3.7.6.1

The CREVS provides for filtration of outside air delivered to the control room by the ventilation system in the event of radioactive releases of particulates or iodine from containment following an accident involving fuel failures. This is to assure that control room personnel are protected from potential radiation exposures in excess of regulatory limits. The system may also provide protection of control room personnel from chemical or toxic gas releases by isolating the control room air intakes.

Currently TS 3.7.6.1 allows a 1-hour CT when both trains are inoperable except when due specifically to an inoperable boundary. In that case a 90 day completion time is allowed after immediately implementing mitigating actions and verifying within 24 hours that the mitigating actions ensure occupant exposures to hazards will not exceed limits. The proposed change would provide a 24-hour CT to restore at least one train of the CREVS to operable status to permit continued operation under an existing action requirement.

In the event of an accident involving radioactive releases without the availability of the CREVS, there would be no direct impact on the capability of the control room staff to perform any actions required to mitigate severe core damage or large early releases, because alternative protective measures would be implemented to reduce the dose impacts. If the accident did not involve severe core damage, control room doses even without the CREVS would be minimal, and therefore the CREVS has no direct role in preventing core damage (i.e., ΔCDF = 0). If a core damage accident did occur with CREVS unavailable, then the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent a release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the CREVS unavailable is assumed to proceed to a large early release, would be:

$$(1E\text{-}4/\text{year}) \times (24 \text{ hours}) \times (\text{year}/8760 \text{ hours}) = 2.7E\text{-}7$$

Assuming a once per three year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated ΔLERF is about 9.0E-8/year. This ΔLERF, and the zero ΔCDF, are below the acceptance guidelines of RG 1.174.

A significant contributor to control room radiological hazards was identified in TR WCAP-16125 as the release of radioactive RCS fluid from a SGTR event. A required TS action will be included in the new proposed action to verify that LCO 3.4.8, "RCS Specific Activity," is met providing additional defense-in-depth.

The TR WCAP-16125 also addressed a TS action to require initiation of mitigating actions to lessen the effects of potential hazards of smoke, chemical, radiological, or toxic gas releases. The NRC staff considered the specific hazards and compensatory measures to be plant-specific, and did not find sufficient information to conclude that the proposed changes are acceptable for these events without a plant-specific evaluation. The RAI response (References 12 & 13) identifies that these mitigating actions were previously reviewed and approved by the NRC staff for Traveler TSTF-448 (Reference 10). The TSTF-448 authorizes a generic TS change to permit a 24-hour CT when the control room boundary is inoperable, and includes the same mitigating actions to assure protection of the control room staff from non-radiological hazards.

Currently TS 3.7.6.1, Action b., applies when two CREVS trains are inoperable due to any reason other than an inoperable control room boundary in all Modes and during movement of recently irradiated fuel assemblies and requires restoration of one train within one hour or to initiate a default shutdown. The TR WCAP-161254 justifies a 24-hour CT for two CREVS trains inoperable for any reason provided that mitigating actions are implemented immediately and it is verified that LCO 3.4.8, "Reactor Coolant System Specific Activity," is met within 1 hour. The ACTIONs are proposed to be revised into a tabular format. In the new format IE condition b. now has three RAs which include:

- Action to initiate mitigating actions immediately,
- Verification that LCO 3.4.8, "Reactor Coolant System Specific Activity," is met within 1 hour, and
- Restoration of at least one CREVS train to operable status within 24 hours.

Additionally, each action contains a default shutdown statement that requires entering Mode 3 in 6 hours and Mode 5 in 30 hours.

Based on the risk result being below the acceptance guidelines of RG 1.174 and the additional restriction on meeting RCS specific activity limits in the TSs, the NRC staff finds the proposed new action requirements and 24-hour CT to be acceptable.

3.4 TS Bases Changes

The TSTF-426 included and the licensee submitted the following TS Bases changes:

- A reference to the NRC-approved TR WCAP-16125 has been added to the reference section of the TS Bases for each TS affected in TSTF-426.
- Revisions to reflect the proposed changes to the TS.
- For all affected TSs, a Note on each applicable condition was added that states: "Not applicable when second [system or component name] intentionally made inoperable." The Bases are revised to provide additional explanation of the Note: "The Condition is modified by a Note stating it is not applicable if the second [system or component name] is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one [system or component name] is inoperable for any reason and the second [system or component name] is discovered to be inoperable, or if both [system or component name] are discovered to be inoperable at the same time."

The NRC staff determined that TS Bases changes in TSTF-426, Revision 5, are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's Final Policy Statement on TSs Improvements for U.S. Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132).

3.5 NRC Staff Findings

The NRC staff has reviewed the licensee's proposed changes in TS content against the changes to STS made by approved Traveler TSTF-426, Revision 5, which was based on approved TR WCAP-16125 (using the five key principles of risk-informed decision making) and concludes that the proposed changes are acceptable. Appropriate TS notes are provided which assure that the loss of safety function action requirements are not applicable for operational convenience and that voluntary entry into these action requirements in lieu of other alternatives that would not result in redundant systems or components being inoperable are prohibited.

The NRC staff further notes that the proposed change does not alter the regulations for notifications and reports required by 10 CFR Part 50 involving the loss of safety function, and that any plant-specific license amendment which provides a condition to address a loss of safety function would not obviate the requirement for a licensee to provide such notifications and reports.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified on June 26, 2015, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase 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 published in the FR on December 23, 2014 (79 FR 77044). Accordingly, the amendment meets the eligibility criteria for categorical exclusion 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 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) there is reasonable assurance that 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. Letter from Dominion to USNRC, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 2, Licensee Amendment Request to Revise Technical Specifications to Adopt TSTF-426, "Revise or Add Actions to Preclude Entry Into LCO 3.0.3 – RITSTF Initiatives 6B & 6C," June 30, 2014 (ADAMS Accession No. ML14188B189).

2. Letter from Dominion to USNRC, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 2, Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Adopt TSTF-426, "Revise or Add Actions to Preclude Entry Into LCO 3.0.3 – RITSTF Initiatives 6B & 6C," January 29, 2015 (ADAMS Accession No. ML14188B189).

3. Letter from Excel Services Corporation, PWR and BWR Owners Group to USNRC, "Transmittal of TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 –RITSTF Initiatives 6b & 6c," November 22, 2011 (ADAMS Accession Number ML113260461).

4. USNRC, NUREG-1432, Revision 4, "Standard Technical Specifications: Combustion Engineering Plants," Volumes 1 & 2, Revision 4, April 2012 (ADAMS Accession No. ML12100A222).

5. USNRC, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," USNRC, May 2011. (ADAMS Accession No. ML100910006).

6. USNRC, NUREG-0800, Standard Review Plan, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (ADAMS Accession No. ML071700658).

7. Letter from Westinghouse Electric Company, PWR and BWR Owners Group to USNRC, Submittal of TR WCAP-16125-NP-A, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," January 5, 2011 (ADAMS Package Accession No. ML110070498).

8. Letter to Mr. Nowinowski, Westinghouse Electric Company from USNRC transmitting "Final Safety Evaluation of Pressurized Water Reactor Owners' Group TR WCAP-16125-NP, Revision 2, "Justification For Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," May 24, 2010 (ADAMS Accession No. ML093560466).

9. USNRC, NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987 – 1995," February 1999 (ADAMS Accession No. ML070580080).

10. NUREG/CR-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," March 1998 (ADAMS Accession No. ML070570094).

11. NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," February 1996 (ADAMS Accession No. ML081920672).

12. Responses to the NRC RAI on TR WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," August 10, 2009 (ADAMS Accession No. ML092260399).

13. Responses to the NRC RAI #2 on TR WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," July 8, 2009 (ADAMS Accession No. ML091940063).

14. Letter from Excel Services Corporation, PWR and BWR Owners Group to USNRC, Transmittal of "TSTF-448-A, Revision 3, "Control Room Habitability," dated August 8, 2006, and corrected pages," December 29, 2006 (ADAMS Accession Nos. ML062210095 and ML063630467).

Principal Contributor: P. Snyder

Date: July 29, 2015

July 29, 2015

Mr. David A. Heacock
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Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISE OR ADD ACTIONS TO PRECLUDE ENTRY INTO LIMITING CONDITION FOR OPERATION 3.0.3 CONSISTENT WITH TSTF-426, REVISION 5 (TAC NO. MF4471)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No. 321 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated June 30, 2014, as supplemented by letter dated January 29, 2015.

The amendment revises the Technical Specifications (TSs) by adopting approved Technical Specification Task Force (TSTF) traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 – RITSTF Initiatives 6b and 6c," and providing a short completion time to restore an inoperable system for conditions under which the previous TS required a plant shutdown. The amendment also reformats each proposed MPS2 custom TS ACTION format to a two-column tabular format.

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/
Richard V. Guzman, Sr. Project Manager
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 321 to DPR-65
2. Safety Evaluation

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ADAMS Accession No.: ML15187A326 * See memo dated June 15, 2015

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DATE	7/06/2015	7/09/2015	6/15/2015	7/28/2015
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DATE	7/28/2015	7/29/2015	7/29/2015	

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