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

July 15, 2009

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION REQUIREMENTS RELATED TO SNUBBERS (TAC NO. MD9337)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 179 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 30, 2008, as supplemented by letter dated February 6, 2009. The amendment: (1) relocates TS 3/4.7.5, "Snubbers," to the HCGS Technical Requirements Manual (TRM); (2) relocates TS 6.10.3.l, which specifies retention requirements for records of snubber service life monitoring, to the TRM; (3) adds new TS Limiting Condition for Operation (LCO) 3.0.8, "Inoperability of Snubbers;" and (4) modifies LCO 3.0.1 to reference LCO 3.0.8.

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

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 179 to License No. NPF-57
2. Safety Evaluation

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

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. NPF-57

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

   A. The application for amendment filed by PSEG Nuclear LLC dated July 30, 2008, as supplemented by letter dated February 6, 2009, 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 179, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days. Implementation of the amendment shall include the relocation of information from the Technical Specifications to the Technical Requirements Manual as described in the licensee's application dated July 30, 2008. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated February 6, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: July 15, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 179

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

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

Remove	Insert
Page 3	Page 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
xiii	xiii
xx	xx
3/4 0-1	3/4 0-1
3/4 0-2	3/4 0-2
3/4 0-3	3/4 0-3
- - -	3/4 0-4
3/4 7-13	3/4 7-13
3/4 7-14	- - -
3/4 7-15	- - -
3/4 7-16	- - -
3/4 7-17	- - -
3/4 7-17a	- - -
3/4 7-17b	- - -
3/4 7-18	- - -
6-22	6-22

(4) PSEG Nuclear LLC, 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;

(5) PSEG Nuclear LLC, 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 calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 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 license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to 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

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 179, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.7.3	FLOOD PROTECTION	3/4 7-9
	Table 3.7.3-1 Perimeter Flood Doors	3/4 7-10
3/4.7.4	REACTOR CORE ISOLATION COOLING SYSTEM	3/4 7-11
3/4.7.5	DELETED	
3/4.7.6	SEALED SOURCE CONTAMINATION	3/4 7-19
3/4.7.7	MAIN TURBINE BYPASS SYSTEM	3/4 7-21
3/4.8	ELECTRICAL POWER SYSTEMS	
3/4.8.1	A.C. SOURCES	
	A.C. Sources-Operating	3/4 8-1
	Table 4.8.1.1.2-1 Diesel Generator Test Schedule	3/4 8-10
	A.C. Sources-Shutdown	3/4 8-11
3/4.8.2	D.C. SOURCES	
	D.C. Sources-Operating	3/4 8-12
	Table 4.8.2.1-1 Battery Surveillance Requirements	3/4 8-15
	D.C. Sources Shutdown	3/4 8-17
3/4.8.3	ONSITE POWER DISTRIBUTION SYSTEMS	
	Distribution - Operating	3/4 8-18
	Distribution - Shutdown	3/4 8-21
3/4.8.4	ELECTRICAL EQUIPMENT PROTECTIVE DEVICES	
	Primary Containment Penetration Conductor Overcurrent Protective Devices	3/4 8-24
	Table 3.8.4.1-1 Primary Containment Penetration Conductor Overcurrent Protective Devices	3/4 8-26
	Motor Operated Valve Thermal Overload Protection (Bypassed)	3/4 8-30

INDEX

BASES

SECTION		PAGE
3/4.7 PLANT SYSTEMS		
3/4.7.1	SERVICE WATER SYSTEMS	B 3/4 7-1
3/4.7.2	CONTROL ROOM EMERGENCY FILTRATION SYSTEM	B 3/4 7-1
3/4.7.3	FLOOD PROTECTION	B 3/4 7-1
3/4.7.4	REACTOR CORE ISOLATION COOLING SYSTEM	B 3/4 7-1a
3/4.7.5	DELETED	
3/4.7.6	SEALED SOURCE CONTAMINATION	B 3/4 7-4
3/4.7.7	MAIN TURBINE BYPASS SYSTEM	B 3/4 7-4
3/4.8 ELECTRICAL POWER SYSTEMS		
3/4.8.1, 3/4.8.2 and 3/4.8.3	A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS	B 3/4 8-1
3/4.8.4	ELECTRICAL EQUIPMENT PROTECTIVE DEVICES	B 3/4 8-3
3/4.9 REFUELING OPERATIONS		
3/4.9.1	REACTOR MODE SWITCH	B 3/4 9-1
3/4.9.2	INSTRUMENTATION	B 3/4 9-1
3/4.9.3	CONTROL ROD POSITION	B 3/4 9-1
3/4.9.4	DELETED	B 3/4 9-1
3/4.9.5	DELETED	B 3/4 9-1
3/4.9.6	DELETED	B 3/4 9-2
3/4.9.7	DELETED	B 3/4 9-2
3/4.9.8 and 3/4.9.9	WATER LEVEL - REACTOR VESSEL and WATER LEVEL - SPENT FUEL STORAGE POOL	B 3/4 9-2
3/4.9.10	CONTROL ROD REMOVAL	B 3/4 9-2
3/4.9.11	RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION	B 3/4 9-2

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, and except as provided in LCO 3.0.8.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL CONDITION or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 Not used.

3.0.7 Not used.

LIMITING CONDITION FOR OPERATION (Continued)

3.0.8 Inoperability of Snubbers

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be a failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within its specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50 Sections 50.55a(f) and 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(f)(6)(i) or Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and

Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall conform to the staff positions on schedule, methods, and personnel, and sample expansion included in that generic letter, or as otherwise approved by the NRC.

3/4.7.5 DELETED

Pages 3/4 7-14 through 3/4 7-18 have been intentionally omitted.

RECORD RETENTION (Continued)

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Quality Assurance Program.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff).

l. DELETED

m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATIONAL MANUAL and the PROCESS CONTROL PROGRAM.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By application dated July 30, 2008, as supplemented by letter dated February 6, 2009, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML082200316 and ML090560538, respectively) PSEG Nuclear LLC (PSEG or the licensee) requested changes to the Technical Specifications (TSs) for the Hope Creek Generating Station (HCGS). The supplement dated February 6, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards determination as published in the *Federal Register* on October 7, 2008 (73 FR 58677).

The application included a related request for relief from certain requirements specified in the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants for the third inservice inspection interval at HCGS. The Nuclear Regulatory Commission (NRC or the Commission) is reviewing the relief request (HC-I3R-04) separately from the proposed license amendment request.

The proposed amendment would: (1) relocate TS 3/4.7.5, "Snubbers," to the HCGS Technical Requirements Manual (TRM); (2) relocate TS 6.10.3.I, which specifies retention requirements for records of snubber service life monitoring, to the TRM; (3) add new TS Limiting Condition for Operation (LCO) 3.0.8, "Inoperability of Snubbers;" and (4) modify LCO 3.0.1 to reference LCO 3.0.8.

The licensee's application dated July 30, 2008, stated that relocating the snubber TS requirements to the TRM would allow PSEG to revise snubber testing requirements in accordance with Section 50.59 of Title 10 of the *Code of Federal Regulations* (10 CFR). The licensee's application also stated that the proposed addition of LCO 3.0.8 (which would establish conditions under which TS systems would remain operable when required snubbers are not capable of providing their related support function) is consistent with NRC-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-372, Revision 4. The NRC staff published a notice of this TS improvement in the *Federal Register* on May 4, 2005 (70 FR 23252) as part of the Consolidated Line Item Improvement Process (CLIIP). The notice included a model safety evaluation (SE) that may be referenced by

Enclosure

licensees in plant-specific applications to adopt the TSTF-372 changes. In its application, the licensee stated that the justifications included in the model SE for TSTF-372 are applicable to HCGS and justify the proposed addition of LCO 3.0.8. The SE that follows is based on the model SE with changes as necessary to address the specific changes proposed for HCGS not included within the scope of TSTF-372 (e.g., relocation of snubber TS requirements to TRM).

On April 23, 2004, the Nuclear Energy Institute (NEI) Risk Informed Technical Specifications Task Force submitted a proposed change, TSTF-372, Revision 4, to the standard technical specifications (STS) (NUREGs 1430-1434) on behalf of the industry (TSTF-372, Revisions 1 through 3 were prior draft iterations). TSTF-372, Revision 4, is a proposal to add an STS Limiting Condition for Operation (LCO) 3.0.8, allowing a delay time for entering a supported system TS, when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges.

This proposal is one of the industry's initiatives being developed under the risk-informed TSs program. These initiatives are intended to maintain or improve safety through the incorporation of risk assessment and management techniques in TS, while reducing unnecessary burden and making TS requirements consistent with the Commission's other risk-informed regulatory requirements, in particular the Maintenance Rule.

The proposed change adds a new limiting condition for operation, LCO 3.0.8, to the TSs. LCO 3.0.8 allows licensees to delay declaring an LCO not met for equipment, supported by snubbers unable to perform their associated support functions, when risk is assessed and managed. This new LCO 3.0.8 states:

> When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:
>
> a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or
>
> b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.
>
> At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

The proposed TS changes are further described in Section 2.0. The technical evaluation and approach used to assess its risk impact is discussed in Section 3.0 with the specific results and insights of the risk assessment discussed in Section 3.1. Section 3.2 summarizes the NRC staff's conclusions from the review of the proposed TS change.

2.0 REGULATORY EVALUATION

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2)(i), "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications..." TS Section 3.0, on "LCO and SR Applicability," provides details or ground rules for complying with the LCOs.

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement) which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953, July 19, 1995). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

> Criterion 1
> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 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.
>
> Criterion 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.
>
> Criterion 4
> A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As discussed in the *Federal Register* notice for the final rule dated July 19, 1995 (60 FR 36955):

> LCOs that do not meet any of the criteria, and their associated actions and surveillance requirements, may be proposed for relocation from the technical specifications to licensee-controlled documents, such as the FSAR [Final Safety Analysis Report]. The criteria may be applied to either standard or custom technical specifications.

Snubbers are chosen in lieu of rigid supports in areas where restricting thermal growth during normal operation would induce excessive stresses in the piping nozzles or other equipment. Although they are classified as component standard supports, they are not designed to provide any transmission of force during normal plant operations. However, in the presence of dynamic transient loadings, which are induced by seismic events as well as by plant accidents and transients, a snubber functions as a rigid support. The location and size of the snubbers are determined by stress analyses based on different combinations of load conditions, depending on the design classification of the particular piping.

Prior to the conversion to the improved STS, TS requirements applied directly to snubbers. These requirements included:

- A requirement that snubbers be functional and in service when the supported equipment is required to be operable;
- A requirement that snubber removal for testing be done only during plant shutdown;
- A requirement that snubber removal for testing be done on a one-at-a-time basis when supported equipment is required to be operable during shutdown;
- A requirement to repair or replace within 72 hours any snubbers, found to be inoperable during operation in Modes 1 through 4, to avoid declaring any supported equipment inoperable;
- A requirement that each snubber be demonstrated operable by periodic visual inspections; and
- A requirement to perform functional tests on a representative sample of at least 10% of plant snubbers, at least once every 18 months during shutdown.

In the late 1980s, a joint initiative of the NRC and industry was undertaken to improve the STS (Reference 10). This effort identified the snubbers as candidates for relocation to a licensee-controlled document based on the fact that the TS requirements for snubbers did not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii) for inclusion in the improved STS. The NRC approved the relocation without placing any restriction on the use of the relocated requirements. However, this relocation resulted in different interpretations between the NRC and the industry regarding its implementation.

The NRC has stated (Reference 11), that since snubbers are supporting safety equipment that is in the TSs, the definition of OPERABILITY must be used to immediately evaluate equipment

supported by a removed snubber and, if found inoperable, the appropriate TS required actions must be entered. This interpretation has, in practice, eliminated the 72-hour delay to enter the actions for the supported equipment that existed prior to the conversion to the improved STS (the only exception is if the supported system has been analyzed and determined to be OPERABLE without the snubber). The industry has argued that since the NRC approved the relocation without placing any restriction on the use of the relocated requirements, the licensee-controlled document requirements for snubbers should be invoked before the supported system's TS requirements become applicable. The industry's interpretation would, in effect, restore the 72-hour delay to enter the actions for the supported equipment that existed prior to the conversion to the improved STS. The industry's proposal would allow a time delay for all conditions, including snubber removal for testing at power.

The option to relocate the snubbers to a licensee-controlled document, as part of the conversion to improved STS, has resulted in non-uniform and inconsistent treatment of snubbers. On the one hand, plants that have relocated snubbers from their TSs are allowed to change the TS requirements for snubbers under the auspices of 10 CFR 50.59, but they are not allowed a 72-hour delay before they enter the actions for the supported equipment. On the other hand, plants that have not converted to improved STS have retained the 72-hour delay if snubbers are found to be inoperable, but they are not allowed to use 10 CFR 50.59 to change TS requirements for snubbers. It should also be noted that a few plants that converted to the improved STS chose not to relocate the snubbers to a licensee-controlled document and, thus, retained the 72-hour delay. In addition, it is important to note that unlike plants that have not relocated, plants that have relocated can perform functional tests on the snubbers at power (as long as they enter the actions for the supported equipment) and at the same time can reduce the testing frequency (as compared to plants that have not relocated) if it is justified by 10 CFR 50.59 assessments. Some potential undesirable consequences of this inconsistent treatment of snubbers are:

- Performance of testing during crowded time period windows when the supported system is inoperable with the potential to reduce the snubber testing to a minimum since the snubber requirements that have been relocated from TSs are controlled by the licensee;

- Performance of testing during crowded windows when the supported system is inoperable with the potential to increase the unavailability of safety systems; and

- Performance of testing and maintenance on snubbers affecting multiple trains of the same supported system during the 7 hours allotted before entering MODE 3 under LCO 3.0.3.

The TSTF proposed a risk-informed TS change that introduces a delay time before entering the actions for the supported equipment, when one or more snubbers are found inoperable or removed for testing, if risk is assessed and managed. Such a delay time will provide needed flexibility in the performance of maintenance and testing during power operation and at the same time will enhance overall plant safety by:

- Avoiding unnecessary unscheduled plant shutdowns and, thus, minimizing plant transition and realignment risks;

- Performing most of the required testing and maintenance during the delay time when the supported system is available to mitigate most challenges and, thus, avoiding increases in safety system unavailability; and

- Providing explicit risk-informed guidance in areas in which that guidance currently does not exist, such as the treatment of snubbers impacting more than one redundant train of a supported system.

3.0 TECHNICAL EVALUATION

The industry submitted TSTF-372, Revision 4, "Addition of LCO 3.0.8, Inoperability of Snubbers" in support of the proposed TS change. This submittal (Reference 1) documents a risk-informed analysis of the proposed TS change. Probabilistic risk assessment (PRA) results and insights are used, in combination with deterministic and defense-in-depth arguments, to identify and justify delay times for entering the actions for the supported equipment associated with inoperable snubbers at nuclear power plants. This is in accordance with guidance provided in Regulatory Guides (RGs) 1.174 and 1.177 (References 2 and 3, respectively).

The risk impact associated with the proposed delay times for entering the TS actions for the supported equipment can be assessed using the same approach as for allowed completion time (CT) extensions. Therefore, the risk assessment was performed following the three-tiered approach recommended in RG 1.177 for evaluating proposed extensions in currently allowed CTs:

- The first tier involves the assessment of the change in plant risk due to the proposed TS change. Such risk change is expressed (1) by the change in the average yearly core damage frequency (ΔCDF) and the average yearly large early release frequency (ΔLERF); and (2) by the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP). The assessed ΔCDF and ΔLERF values are compared to acceptance guidelines, consistent with the Commission's Safety Goal Policy Statement as documented in RG 1.174, so that the plant's average baseline risk is maintained within a minimal range. The assessed ICCDP and ICLERP values are compared to acceptance guidelines provided in RG 1.177, which aim at ensuring that the plant risk does not increase unacceptably during the period the equipment is taken out of service.

- The second tier involves the identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk-significant operational factors such as concurrent equipment testing were also involved. The objective is to ensure that appropriate restrictions are in place to avoid any potential high-risk configurations.

- The third tier involves the establishment of an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified. The objective of the CRMP is to manage configuration-specific risk by appropriate scheduling of plant activities and/or appropriate compensatory measures.

A simplified bounding risk assessment was performed to justify the proposed addition of LCO 3.0.8 to the TSs. This approach was necessitated by (1) the general nature of the proposed TS changes (i.e., they apply to all plants and are associated with an undetermined number of snubbers that are not able to perform their function); (2) the lack of detailed engineering analyses that establish the relationship between earthquake level and supported system pipe failure probability when one or more snubbers are inoperable; and (3) the lack of seismic risk assessment models for most plants. The simplified risk assessment is based on the following major assumptions, which the staff finds acceptable, as discussed below:

- The accident sequences contributing to the risk increase associated with the proposed TS changes are assumed to be initiated by a seismically-induced loss-of-offsite-power (LOOP) event with concurrent loss of all safety system trains supported by the out-of-service snubbers. In the case of snubbers associated with more than one train (or subsystem) of the same system, it is assumed that all affected trains (or subsystems) of the supported system are failed. This assumption was introduced to allow the performance of a simple bounding risk assessment approach with application to all plants. This approach was selected due to the lack of detailed plant-specific seismic risk assessments for most plants and the lack of fragility data for piping when one or more supporting snubbers are inoperable.

- The LOOP event is assumed to occur due to the seismically-induced failure of the ceramic insulators used in the power distribution systems. These ceramic insulators have a high confidence (95%) of low probability (5%) of failure (HCLPF) of about 0.1g, expressed in terms of peak ground acceleration. Thus, a magnitude 0.1g earthquake is conservatively assumed to have 5% probability of causing a LOOP initiating event. The fact that no LOOP events caused by higher magnitude earthquakes were considered is justified because (1) the frequency of earthquakes decreases with increasing magnitude; and (2) historical data (References 4 and 5) indicate that the mean seismic capacity of ceramic insulators (used in seismic PRAs), in terms of peak ground acceleration, is about 0.3g, which is significantly higher than the 0.1g HCLPF value. Therefore, the simplified analysis, even though it does not consider LOOP events caused by earthquakes of magnitude higher than 0.1g, bounds a detailed analysis which would use mean seismic failure probabilities (fragilities) for the ceramic insulators.

- Analytical and experimental results obtained in the mid-1980s as part of the industry's "Snubber Reduction Program" (References 4 and 6) indicated that piping systems have large margins against seismic stress. The assumption that a magnitude 0.1g earthquake would cause the failure of all safety system trains supported by the out-of-service snubbers is very conservative because safety piping systems could withstand much higher seismic stresses even when one or more supporting snubbers are out of service. The actual piping failure probability is a function of the stress allowable and the number of snubbers removed for maintenance or testing. Since the licensee controlled testing is done on only a small (about 10%) representative sample of the total snubber population, typically only a few snubbers supporting a given safety system are out for testing at a time. Furthermore, since the testing of snubbers is a planned activity, licensees have flexibility in selecting a sample set of snubbers for testing from a much larger population by conducting configuration-specific engineering and/or risk assessments. Such a selection of snubbers for testing provides confidence that the supported systems would

perform their functions in the presence of a design-basis earthquake and other dynamic loads and, in any case, the risk impact of the activity will remain within the limits of acceptability defined in risk-informed RGs 1.174 and 1.177.

- The analysis assumes that one train (or subsystem) of all safety systems is unavailable during snubber testing or maintenance (an entire system is assumed unavailable if a removed snubber is associated with both trains of a two-train system). This is a very conservative assumption for the case of corrective maintenance since it is unlikely that a visual inspection will reveal that one or more snubbers across all supported systems are inoperable. This assumption is also conservative for the case of the licensee-controlled testing of snubbers since such testing is performed only on a small representative sample.

- In general, no credit is taken for recovery actions and alternative means of performing a function, such as the function performed by a system assumed failed (e.g., when LCO 3.0.8b applies). However, most plants have reliable alternative means of performing certain critical functions. For example, feed and bleed (F&B) can be used to remove heat in most pressurized water reactors (PWRs) when auxiliary feedwater (AFW), the most important system in mitigating LOOP accidents, is unavailable. Similarly, if high pressure makeup (e.g., reactor core isolation cooling) and heat removal capability (e.g., suppression pool cooling) are unavailable in boiling water reactors (BWRs), reactor depressurization in conjunction with low pressure makeup (e.g., low pressure coolant injection) and heat removal capability (e.g., shutdown cooling) can be used to cool the core. A 10% failure probability for recovery actions to provide core cooling using alternative means is assumed for Diablo Canyon, the only West Coast PWR plant with F&B capability, when a snubber impacting more than one train of the AFW system (i.e., when LCO 3.0.8b is applicable) is out of service. This failure probability value is significantly higher than the value of 2.2E-2 used in Diablo Canyon's PRA. Furthermore, Diablo Canyon has analyzed the impact of a single limiting snubber failure, and concluded that no single snubber failure would impact two trains of the AFW system. No credit for recovery actions to provide core cooling using alternative means is necessary for West Coast PWR plants with no F&B capability because it has been determined that there is no single snubber whose non-functionality would disable two trains of the AFW system in a seismic event of magnitude up to the plant's safe shutdown earthquake (SSE). It should be noted that a similar credit could have been applied to most Central and Eastern US plants but this was not necessary to demonstrate the low risk impact of the proposed TS change due to the lower earthquake frequencies at Central and Eastern US plants as compared to West Coast plants.

- The earthquake frequency at the 0.1g level was assumed to be 1E-3/year for Central and Eastern US plants and 1E-1/year for West Coast plants. Each of these two values envelop the range of earthquake frequency values at the 0.1g level, for Eastern US and West Coast sites, respectively (References 5 and 7).

- The risk impact associated with non-LOOP accident sequences (e.g., seismically initiated loss-of-coolant-accident (LOCA) or anticipated-transient-without-scram (ATWS) sequences) was not assessed. However, this risk impact is small compared to the risk impact associated with the LOOP accident sequences modeled in the simplified

bounding risk assessment. Non-LOOP accident sequences, due to the ruggedness of nuclear power plant designs, require seismically-induced failures that occur at earthquake levels above 0.3g. Thus, the frequency of earthquakes initiating non-LOOP accident sequences is much smaller than the frequency of seismically-initiated LOOP events. Furthermore, because of the conservative assumption made for LOOP sequences that a 0.1g level earthquake would fail all piping associated with inoperable snubbers, non-LOOP sequences would not include any more failures associated with inoperable snubbers than LOOP sequences. Therefore, the risk impact of inoperable snubbers associated with non-LOOP accident sequences is small compared to the risk impact associated with the LOOP accident sequences modeled in the simplified bounding risk assessment.

- The risk impact of dynamic loadings other than seismic loads is not assessed. These shock-type loads include thrust loads, blowdown loads, waterhammer loads, steamhammer loads, LOCA loads and pipe rupture loads. However, there are some important distinctions between non-seismic (shock-type) loads and seismic loads which indicate that, in general, the risk impact of the out-of-service snubbers is smaller for non-seismic loads than for seismic loads. First, while a seismic load affects the entire plant, the impact of a non-seismic load is localized to a certain system or area of the plant. Second, although non-seismic shock loads may be higher in total force and the impact could be as much or more than seismic loads, generally they are of much shorter duration than seismic loads. Third, the impact of non-seismic loads is more plant specific, and thus is harder to analyze generically, than is the impact of seismic loads. For these reasons, licensees will be required to confirm every time LCO 3.0.8 is used, that at least one train of each system that is supported by the inoperable snubber(s) would remain capable of performing the system's required safety or support functions for postulated design loads other than seismic loads.

3.1 Risk Assessment Results and Insights

The results and insights from the implementation of the three-tiered approach of RG 1.177 to support the proposed addition of LCO 3.0.8 to the TSs are summarized and evaluated in the following Sections 3.1.1 to 3.1.3.

3.1.1 Risk Impact

The bounding risk assessment approach, discussed in Section 3.0, was implemented generically for all U.S. operating nuclear power plants. Risk assessments were performed for two categories of plants, Central and East Coast plants and West Coast plants, based on historical seismic hazard curves (earthquake frequencies and associated magnitudes). The first category, Central and East Coast plants, includes the vast majority of the U.S. nuclear power plant population (Reference 7). For each category of plants, two risk assessments were performed:

- The first risk assessment applies to cases where all inoperable snubbers are associated with only one train (or subsystem) of the impacted safety systems. It was conservatively assumed that a single train (or subsystem) of each safety system is unavailable. It was also assumed that the probability of non-mitigation using the unaffected redundant trains

(or subsystems) is 2%. This is a conservative value given that for core damage to occur under those conditions, two or more failures are required.

- The second risk assessment applies to the case where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety systems. It was assumed in this bounding analysis that all safety systems are unavailable to mitigate the accident, except for West Coast PWR plants. Credit for using F&B to provide core cooling is taken for plants having F&B capability (e.g., Diablo Canyon) when a snubber impacting more than one train of the AFW system is inoperable. Credit for one AFW train to provide core cooling is taken for West Coast PWR plants with no F&B capability (e.g., San Onofre) because it has been determined that there is no single snubber whose non-functionality would disable two trains of the AFW system in a seismic event of magnitude up to the plant's SSE.

The results of the performed risk assessments, in terms of core damage and large early release risk impacts, are summarized in Table 1. The first row lists the conditional risk increase, in terms of CDF (core damage frequency), ΔR_{CDF}, caused by the out-of-service snubbers (as assumed in the bounding analysis). The second and third rows list the ICCDP (incremental conditional core damage probability) and the ICLERP (incremental conditional large early release probability) values, respectively. The ICCDP for the case where all inoperable snubbers are associated with only one train (or subsystem) of the supported safety systems, was obtained by multiplying the corresponding ΔR_{CDF} value by the time fraction of the proposed 72-hour delay to enter the actions for the supported equipment. The ICCDP for the case where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety system, was obtained by multiplying the corresponding ΔR_{CDF} value by the time fraction of the proposed 12-hour delay to enter the actions for the supported equipment. The ICLERP values were obtained by multiplying the corresponding ICCDP values by 0.1 (i.e., by assuming that the ICLERP value is an order of magnitude less than the ICCDP). This assumption is conservative since containment bypass scenarios, such as steam generator tube rupture accidents and interfacing system LOCAs, would not be uniquely affected by the out-of-service snubbers. Finally, the fourth and fifth rows list the assessed ΔCDF and ΔLERF values, respectively. These values were obtained by dividing the corresponding ICCDP and ICLERP values by 1.5 (i.e., by assuming that the snubbers are tested every 18 months, as was the case before the snubbers were relocated to a licensee-controlled document). This assumption is reasonable because (1) it is not expected that licensees would test the snubbers more often than what used to be required by the TSs; and (2) testing of snubbers is associated with higher risk impact than the average corrective maintenance of snubbers found inoperable by visual inspection (testing is expected to involve significantly more snubbers out-of-service than corrective maintenance). The assessed ΔCDF and ΔLERF values are compared to acceptance guidelines, consistent with the Commission's Safety Goal Policy Statement as documented in RG 1.174, so that the plant's average baseline risk is maintained within a minimal range. This comparison indicates that the addition of LCO 3.0.8 to the existing TS would have an insignificant risk impact.

Table 1
Bounding Risk Assessment Results for Snubbers Impacting a Single Train and Multiple Trains of a Supported System

	Central and East Coast Plants		West Coast Plants	
	Single Train	Multiple Train	Single Train	Multiple Train
ΔR_{CDF}/yr	1E-6	5E-6	1E-4	5E-4
ICCDP	8E-9	7E-9	8E-7	7E-7
ICLERP	8E-10	7E-10	8E-8	7E-8
ΔCDF/yr	5E-9	5E-9	5E-7	5E-7
ΔLERF/yr	5E-10	5E-10	5E-8	5E-8

The assessed ΔCDF and ΔLERF values meet the acceptance criteria of 1E-6/year and 1E-7/year, respectively, based on guidance provided in RG 1.174. This conclusion is true without taking any credit for the removal of potential undesirable consequences associated with the current inconsistent treatment of snubbers (e.g., reduced snubber testing frequency, increased safety system unavailability and treatment of snubbers impacting multiple trains) discussed in Section 2.0 above, and given the bounding nature of the risk assessment.

The assessed ICCDP and ICLERP values are compared to acceptance guidelines provided in RG 1.177, which aim at ensuring that the plant risk does not increase unacceptably during the period the equipment is taken out of service. This comparison indicates that the addition of LCO 3.0.8 to the existing TSs meets the RG 1.177 numerical guidelines of 5E-7 for ICCDP and 5E-8 for ICLERP. The small deviations shown for West Coast plants are acceptable because of the bounding nature of the risk assessments, as discussed in Section 3.0.

The risk assessment results of Table 1 are also compared to guidance provided in the revised Section 11 of NUMARC 93-01, Revision 2 (Reference 8), endorsed by RG 1.182 (Reference 9), for implementing the requirements of paragraph (a)(4) of the Maintenance Rule, 10 CFR 50.65. Such guidance is summarized in Table 2. Guidance regarding the acceptability of conditional risk increase in terms of CDF (i.e., ΔR_{CDF}) for a planned configuration is provided. This guidance states that a specific configuration that is associated with a CDF higher than 1E-3/year should not be entered voluntarily. Since the assessed conditional risk increase, ΔR_{CDF}, is significantly less than 1E-3/year, plant configurations including out of service snubbers and other equipment may be entered voluntarily if supported by the results of the risk assessment required by 10 CFR 50.65(a)(4), by LCO 3.0.8, or by other TS.

Table 2
Guidance for Implementing 10 CFR 50.65(a)(4)

ΔR_{CDF}	Guidance
Greater than 1E-3/year	Configuration should not normally be entered voluntarily

ICCDP	Guidance	ICLERP
Greater than 1E-5	Configuration should not normally be entered voluntarily	Greater than 1E-6
1E-6 to 1E-5	Assess non-quantifiable factors; Establish risk management actions	1E-7 to 1E-6
Less than 1E-6	Normal work controls	Less than1E-7

Guidance regarding the acceptability of ICCDP and ICLERP values for a specific planned configuration and the establishment of risk management actions is also provided in NUMARC 93-01. This guidance, as shown in Table 2, states that a specific plant configuration that is associated with ICCDP and ICLERP values below 1E-6 and 1E-7, respectively, is considered to require "normal work controls." Table 1 shows that for the majority of plants (i.e., for all plants in the Central and East Coast category) the conservatively assessed ICCDP and ICLERP values are over an order of magnitude less than what is recommended as the threshold for the "normal work controls" region. For West Coast plants, the conservatively assessed ICCDP and ICLERP values are still within the "normal work controls" region. Thus, the risk contribution from out-of-service snubbers is within the normal range of maintenance activities carried out at a plant. Therefore, plant configurations involving out-of-service snubbers and other equipment may be entered voluntarily if supported by the results of the risk assessment required by 10 CFR 50.65(a)(4), by LCO 3.0.8, or by other TSs. However, this simplified bounding analysis indicates that for West Coast plants the provisions of LCO 3.0.8 must be used cautiously and in conjunction with appropriate management actions, especially when equipment other than snubbers is also inoperable, based on the results of configuration-specific risk assessments required by 10 CFR 50.65(a)(4), by LCO 3.0.8, or by other TSs.

The NRC staff finds that the risk assessment results support the proposed addition of LCO 3.0.8 to the TSs. The risk increases associated with this TS change will be insignificant based on guidance provided in RGs 1.174 and 1.177 and within the range of risks associated with normal maintenance activities. In addition, LCO 3.0.8 will remove potential undesirable consequences stemming from the current inconsistent treatment of snubbers in the TSs, such as reduced frequency of snubber testing, increased safety system unavailability and the treatment of snubbers impacting multiple trains.

3.1.2 Identification of High-Risk Configurations

The second tier of the three-tiered approach recommended in RG 1.177 involves the identification of potentially high-risk configurations that could exist if equipment, in addition to that associated with the TS change, were to be taken out of service simultaneously. Insights

from the risk assessments, in conjunction with important assumptions made in the analysis and defense-in-depth considerations, were used to identify such configurations. To avoid these potentially high-risk configurations, specific restrictions to the implementation of the proposed TS changes were identified.

For cases where all inoperable snubbers are associated with only one train (or subsystem) of the impacted systems (i.e., when LCO 3.0.8a applies), it was assumed in the analysis that there will be unaffected redundant trains (or subsystems) available to mitigate the seismically initiated LOOP accident sequences. This assumption implies that there will be at least one success path available when LCO 3.0.8a applies. Therefore, potentially high-risk configurations can be avoided by ensuring that such a success path exists when LCO 3.0.8a applies. Based on a review of the accident sequences that contribute to the risk increase associated with LCO 3.0.8a, as modeled by the simplified bounding analysis (i.e., accident sequences initiated by a seismically-induced LOOP event with concurrent loss of all safety system trains supported by the out of service snubbers), the following restrictions were identified to prevent potentially high-risk configurations:

- For PWR plants, at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), must be available when LCO 3.0.8a is used.

- For BWR plants, one of the following two means of heat removal must be available when LCO 3.0.8a is used:

  - At least one high-pressure makeup path (e.g., using high-pressure coolant injection or reactor core isolation cooling or equivalent) and heat removal capability (e.g., suppression pool cooling), including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s), or

  - At least one low-pressure makeup path (e.g., low-pressure coolant injection or core spray) and heat removal capability (e.g., suppression pool cooling or shutdown cooling), including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s).

For cases where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety system (i.e., when LCO 3.0.8b applies), it was assumed in the bounding analysis that all safety systems are unavailable to mitigate the accident, except for West Coast plants. Credit for using F&B to provide core cooling is taken for plants having F&B capability (e.g., Diablo Canyon) when a snubber impacting more than one train of the AFW system is inoperable. Credit for one AFW train to provide core cooling is taken for West Coast PWR plants with no F&B capability (e.g., San Onofre) because it has been determined that there is no single snubber whose non-functionality would disable more than one train of AFW in a seismic event of magnitude up to the plant's SSE. Based on a review of the accident sequences that contribute to the risk increase associated with LCO 3.0.8b (as modeled by the simplified bounding analysis) and on defense-in-depth considerations, the following restrictions were identified to prevent potentially high-risk configurations:

- LCO 3.0.8b cannot be used at West Coast PWR plants with no F&B capability when a snubber whose non-functionality would disable more than one train of AFW in a seismic event of magnitude up to the plant's SSE is inoperable (it should be noted, however, that based on information provided by the industry, there is no plant that falls in this category),

- When LCO 3.0.8b is used at PWR plants, at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling (e.g., F&B, firewater system or "aggressive secondary cooldown" using the steam generators) must be available, and

- When LCO 3.0.8b is used at BWR plants, it must be verified that at least one success path exists, using equipment not associated with the inoperable snubber(s), to provide makeup and core cooling needed to mitigate LOOP accident sequences.

3.1.3 Configuration Risk Management

The third tier of the three-tiered approach recommended in RG 1.177 involves the establishment of an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified. The objective of the CRMP is to manage configuration-specific risk by appropriate scheduling of plant activities and/or appropriate compensatory measures. This objective is met by licensee programs to comply with the requirements of paragraph (a)(4) of the Maintenance Rule (10 CFR 50.65) to assess and manage risk resulting from maintenance activities, and by the TSs requiring risk assessments and management using (a)(4) processes if no maintenance is in progress. These programs can support licensee decision-making regarding the appropriate actions to manage risk whenever a risk-informed TS is entered. Since the 10 CFR 50.65(a)(4) guidance (Reference 8) does not currently address seismic risk, licensees adopting this change must ensure that the proposed LCO 3.0.8 is considered with respect to other plant maintenance activities and integrated into the existing 10 CFR 50.65(a)(4) process whether the process is invoked by a TS or by (a)(4) itself.

3.1.4 Relocated Requirements

As discussed above in Section 2.0, LCOs that do not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and their associated actions and surveillance requirements, may be relocated from the TSs to a licensee-controlled document. The licensee has proposed to relocate HCGS TS 3/4.7.5, "Snubbers," to the HCGS TRM. As discussed in the licensee's application dated July 30, 2008, its letter dated August 23, 2007 (ADAMS Accession No. ML072480611) and Section 13.5.4 of the HCGS Updated Final Safety Analysis Report (UFSAR), changes to the TRM are controlled in a manner consistent with procedures fully or partially described in the UFSAR and are subject to the provisions of 10 CFR 50.59.

The licensee, in its application dated July 30, 2008, provided the following discussion of the criteria in 10 CFR 50.36(c)(2)(ii) to justify the relocation of TS 3/4.7.5 to the TRM:

1. Snubbers are not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the

reactor coolant pressure boundary. Therefore, the HCGS snubbers do not satisfy Criterion 1.

2. Snubbers are design features used to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient. However, the snubbers are not explicitly considered in the accident analysis and are not considered a required initial condition for a design basis accident or transient to maintain the integrity of a fission product barrier. The effects of an inoperable snubber will be controlled by the Technical Specification requirements of the supported system. The availability of the snubbers is assured based on the performance of periodic inspections and testing. Therefore, the HCGS snubbers do not satisfy Criterion 2.

3. Safety-related snubbers are design-features that function during accidents or severe transients to prevent the propagation of an event to systems that are part of the primary success path for accident mitigation. However, snubbers are not explicitly considered in the accident analysis, but are a structural design feature whose operation is assured by an inspection program. The snubbers are not part of the primary success path for accident mitigation; therefore the HCGS snubbers do not satisfy Criterion 3.

4. Operational experience or probabilistic safety assessment have not shown snubber performance to be significant to the public health and safety. Therefore, the HCGS snubbers do not satisfy Criterion 4.

The NRC staff has reviewed the justification provided by the licensee and finds that relocation of TS 3/4.7.5 to the TRM is acceptable because (1) its inclusion in the TSs is not required by 10 CFR 50.36 or other NRC regulations; (2) snubber operability and surveillance requirements will be contained in the TRM; and (3) future changes to the relocated material will be adequately controlled under the provisions of 10 CFR 50.59.

HCGS TS 6.10.3.l, which specifies the retention requirements for "Records of the snubber service life monitoring pursuant to Technical Specification 4.7.5" would also be relocated to the TRM. TS 6.10.3 requires that these records be retained "for the duration of the unit Operating License." The licensee's application dated July 30, 2008, provided the following justification for the relocation of TS 6.10.3.l:

> Record retention requirements are also located in 10 CFR [Part] 50, Appendix B, and in 10 CFR 50.71. It is not necessary to include redundant or additional requirements in the TS administrative controls.

Criterion XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50 states, in part, that "[s]ufficient records shall be maintained to furnish evidence of activities affecting quality." In addition, 10 CFR 50.71, "Maintenance of records, making of reports," requires that records required by 10 CFR Part 50 be retained "until the Commission terminates the facility license" if the retention period if not otherwise specified. The NRC staff finds that records of the snubber

service life monitoring is an activity affecting quality and, as such, would be covered under Criterion XVII of Appendix B to 10 CFR Part 50. In addition, the requirements in 10 CFR 50.71 ensure that these records will be retained for same duration as the current TS requirements. Based on the requirements in Appendix B to 10 CFR Part 50 and 10 CFR 50.71, the NRC staff finds that relocation of TS 6.10.3.l to the TRM is acceptable.

The NRC staff notes that removal of TS 6.10.3.l and TS 3/4.7.5 from the TSs is consistent with NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4."

3.2 Summary and Conclusions

Relocating snubbers to a licensee-controlled document, as part of the conversion to improved STS (iSTS), has resulted in non-uniform and inconsistent treatment of operability decisions for systems and components with inoperable snubbers at non-iSTS plants as compared to iSTS plants. This has lead to potential undesirable consequences regarding testing when inoperable snubbers are determined to exist on TS-required systems (as discussed above in Section 2.0).

HCGS, a non-iSTS plant, is proposing a risk-informed TS change which introduces a delay time before entering the actions for the supported equipment when one or more snubbers are found inoperable or removed for testing to be consistent with the format and content for the treatment of snubbers at plants having iSTS. Such a delay time will provide flexibility in the performance of maintenance and testing during power operation and at the same time will enhance overall plant safety by (1) avoiding unnecessary unscheduled plant shutdowns, thus, minimizing plant transition and realignment risks; (2) avoiding reduced snubber testing, thus, increasing the availability of snubbers to perform their supporting function; (3) performing most of the required testing and maintenance during the delay time when the supported system is available to mitigate most challenges, thus, avoiding increases in safety system unavailability; and (4) providing explicit risk-informed guidance in areas in which that guidance currently does not exist, such as the treatment of snubbers impacting more than one redundant train of a supported system.

The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A simplified bounding risk assessment was performed to justify the proposed TS changes. This bounding assessment assumes that the risk increase associated with the proposed addition of LCO 3.0.8 to the TSs is associated with accident sequences initiated by a seismically-induced LOOP event with concurrent loss of all safety system trains supported by the out-of-service snubbers. In the case of snubbers associated with more than one train, it is assumed that all affected trains of the supported system are failed. This assumption was introduced to allow the performance of a simple bounding risk assessment approach with application to all plants and was selected due to the lack of detailed plant-specific seismic risk assessments for most plants and the lack of fragility data for piping when one or more supporting snubbers are inoperable. The impact from the addition of the proposed LCO 3.0.8 to the TSs on defense-in-depth was also evaluated in conjunction with the risk assessment results.

Based on this integrated evaluation, the NRC staff concludes that the proposed addition of LCO 3.0.8 to the TSs would lead to insignificant risk increases, if any. Indeed, this conclusion is true without taking any credit for the removal of potential undesirable consequences associated

with the current inconsistent treatment of snubbers, such as the effects of avoiding a potential reduction in the snubber testing frequency and increased safety system unavailability.

Consistent with the NRC staff's approval and inherent in the implementation of TSTF-372, licensees interested in implementing LCO 3.0.8 must, as applicable, operate in accordance with the following stipulations:

1. Appropriate plant procedures and administrative controls will be used to implement the following Tier 2 Restrictions:

   (a) At least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s) must be available when LCO 3.0.8a is used at PWR plants. This stipulation is not applicable to HCGS since it is a BWR plant.

   (b) At least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling (e.g., F&B, fire water system or "aggressive secondary cooldown" using the steam generators), must be available when LCO 3.0.8b is used at PWR plants. This stipulation is not applicable to HCGS since it is a BWR plant.

   (c) LCO 3.0.8b cannot be used by West Coast PWR plants with no F&B capability when a snubber, whose non-functionality would disable more than one train of AFW in a seismic event of magnitude up to the plant's SSE, is inoperable. This stipulation is not applicable to HCGS since it is a BWR plant.

   (d) BWR plants must verify, every time the provisions of LCO 3.0.8 are used, that at least one success path, involving equipment not associated with the inoperable snubber(s), exists to provide makeup and core cooling needed to mitigate LOOP accident sequences.

   (e) Every time the provisions of LCO 3.0.8 are used licensees will be required to confirm that at least one train (or subsystem) of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. LCO 3.0.8 does not apply to non-seismic snubbers. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall be available on a recoverable basis for staff inspection.

2. When the licensee implements the provisions of LCO 3.0.8 for snubbers, which include delay times to enter the actions for the supported equipment when one or more snubbers are out-of-service for maintenance or testing, it must be done in accordance with an overall CRMP to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are

> identified and avoided, as discussed in the TSTF-372 proposed TS Bases. This objective is met by licensee programs to comply with the requirements of paragraph (a)(4) of the Maintenance Rule, 10 CFR 50.65, to assess and manage risk resulting from maintenance activities or when this process is invoked by LCO 3.0.8 or other TSs. These programs can support licensee decision-making regarding the appropriate actions to manage risk whenever a risk-informed TS is entered. Since the 10 CFR 50.65(a)(4) guidance (the revised (May 2000) Section 11 of NUMARC 93-01), does not currently address seismic risk; licensees adopting this change must ensure that the proposed LCO 3.0.8 is considered in conjunction with other plant maintenance activities and integrated into the existing 10 CFR 50.65(a)(4) process. In the absence of a detailed seismic PRA, a bounding risk assessment, such as utilized in this Safety Evaluation, shall be followed.

PSEG's letter dated February 6, 2009, confirmed that appropriate plant procedures and administrative controls would be used to implement the applicable Tier 2 restrictions. PSEG's letter also stated that, upon implementation of the amendment, the HCGS UFSAR would be revised to describe the applicable Tier 2 restrictions.

PSEG provided the following information in its application dated July 30, 2008, regarding plant-specific verifications and commitments:

> The licensee will establish TS Bases for LCO 3.0.8 which provide guidance and details on how to implement the new requirements. LCO 3.0.8 requires that risk be managed and assessed. The Bases will also state that while the Industry and NRC guidance on implementation of 10 CFR 50.65(a)(4), the Maintenance Rule, does not address seismic risk, LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative assessment of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. Finally, HCGS has a Bases Control Program consistent with Section 5.5 of the STS.

Concerning the applicability of the NRC staff's CLIIP safety evaluation for TSTF-372, PSEG stated in its application dated July 30, 2008, that:

> PSEG has reviewed the safety evaluation dated May 4, 2005 as part of the CLIIP. This review included a review of the NRC staff's evaluation, as well as the supporting information provided to support TSTF-372. PSEG has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to HCGS and justify this amendment for the incorporation of the changes to the HCGS TS.

Based on its own review, the NRC staff agrees that the justifications prepared by the NRC staff for TSTF-372 (as detailed in the above safety evaluation) are applicable to HCGS. In addition, the procedure and administrative controls to implement the applicable Tier 2 restrictions and

revision of the UFSAR to describe the applicable Tier 2 restrictions provides reasonable assurance that the appropriate restrictions will be put in place to avoid potential high-risk configurations at HCGS. Based on these considerations, the NRC staff concludes that the proposed addition of LCO 3.0.8 to the HCGS TSs is acceptable. In addition, PSEG has proposed to make a conforming change to LCO 3.0.1 to reference LCO 3.0.8. This change is consistent with TSTF-372 and, therefore, is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified 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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant 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 (73 FR 58677). 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) 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. TSTF-372, Revision 4, "Addition of LCO 3.0.8, Inoperability of Snubbers," April 23, 2004.

2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," USNRC, August 1998.

3. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," USNRC, August 1998.

4. Budnitz, R. J. et. al., "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," NUREG/CR-4334, Lawrence Livermore National Laboratory, July 1985.

5. Advanced Light Water Reactor Utility Requirements Document, Volume 2, ALWR Evolutionary Plant, PRA Key Assumptions and Groundrules, Electric Power Research Institute, August 1990.

6. Bier V. M. et. al., "Development and Application of a Comprehensive Framework for Assessing Alternative Approaches to Snubber Reduction," International Topical Conference on Probabilistic Safety Assessment and Risk Management PSA '87, Swiss Federal Institute of Technology, Zurich, August 30-September 4, 1987.

7. NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," April 1994.

8. NEI, Revised Section 11 of Revision 2 of NUMARC 93-01, May 2000.

9. Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000.

10. SECY-86-10 dated January 13, 1986, "Recommendations for Improving Technical Specifications."

11. Letter from F. Rinaldi, NRC, to H. Barron, Duke Energy Corporation, "Licensing Position Regarding Snubbers," July 9, 1999.

Principal Contributors: K. Bucholtz
C. Schulten
R. Ennis

Date: July 15, 2009

July 15, 2009

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION REQUIREMENTS RELATED TO SNUBBERS (TAC NO. MD9337)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 179 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 30, 2008, as supplemented by letter dated February 6, 2009. The amendment: (1) relocates TS 3/4.7.5, "Snubbers," to the HCGS Technical Requirements Manual (TRM); (2) relocates TS 6.10.3.l, which specifies retention requirements for records of snubber service life monitoring, to the TRM; (3) adds new TS Limiting Condition for Operation (LCO) 3.0.8, "Inoperability of Snubbers;" and (4) modifies LCO 3.0.1 to reference LCO 3.0.8.

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

Sincerely,
/ra/
Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 179 to License No. NPF-57
2. Safety Evaluation

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