February 13, 2006

Mr. Michael R. Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - ISSUANCE OF AMENDMENTS RE: INOPERABILITY OF SNUBBERS (TAC NOS. MC7310 AND MC7311)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 245 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 and Amendment No. 229 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 8, 2005.

The amendments revise the TSs by 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 consistent with the program in place for complying with the requirements of Title 10 of the *Code of Federal Regulations*, Section 50.65(a)(4). Limiting Condition for Operation 3.0.8 is added to the TSs to provide this allowance and define the requirements and limitations for its use.

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

Sincerely,

/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. Amendment No. 245 to DPR-26
2. Amendment No. 229 to DPR-64
3. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML060120013

OFFICE	CLIIP LPM	LPL1-1/PM	LPL1-1/LA	ITSB	OGC	LPL1-1/BC
NAME	WReckley	JBoska	SLittle	RTjader	SHamrick	RLaufer
DATE	1/10/06	1/23/06	1/26/06	1/31/06	2/7/06	2/8/06

Official Record Copy

DATED: February 13, 2006

AMENDMENT NO. 245 TO FACILITY OPERATING LICENSE NO. DPR-26 INDIAN POINT UNIT 2 AND AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

PUBLIC
LPL1-1 R/F
RidsNrrDorlLpla
RidsNrrDirsItsb
RidsNrrPMJBoska
RidsNrrLASLittle
BMcDermott, RI
GHill (4)
RidsOGCMailCenter
RidsACRSACNWMailCenter
RidsNrrPMWReckley

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ENTERGY NUCLEAR INDIAN POINT 2, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 245
License No. DPR-26

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated June 8, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 245 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 13, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 245

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

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 Pages	Insert Pages
3.0-1	3.0-1
3.0-3	3.0-3

ENTERGY NUCLEAR INDIAN POINT 3, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 229
License No. DPR-64

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated June 8, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 229, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 13, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 229

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

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 Pages	Insert Pages
3.0-1	3.0-1
3.0-3	3.0-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 245 TO FACILITY OPERATING LICENSE NO. DPR-64

AND AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3

DOCKET NOS. 50-247 AND 50-286

1.0 INTRODUCTION

By letter dated June 8, 2005, Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) Technical Specifications (TSs). The proposed changes would add Limiting Condition for Operation (LCO) 3.0.8 to address conditions where one or more snubbers are unable to perform their associated support function. These changes are based on Technical Specification Task Force (TSTF) change traveler TSTF-372, Revision 4 that has been approved generically for the Standard Technical Specifications (STS; NUREGs-1430 - 1434). A notice announcing the availability of this proposed TS change using the consolidated line item improvement process (CLIIP) was published in the *Federal Register* on May 4, 2005 (70 FR 23252).

On April 23, 2004, the Nuclear Energy Institute (NEI) Risk Informed Technical Specifications Task Force (RITSTF) submitted a proposed change, TSTF-372, Revision 4, to the STS 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 LCO 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 technical specifications program. These initiatives are intended to maintain or improve safety through the incorporation of risk assessment and management techniques in TSs; 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 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.

2.0 REGULATORY EVALUATION

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs 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 TS. As stated in 10 CFR 50.36(c)(2)(i), the "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 specification ... ." TS Section 3.0, on "LCO and SR Applicability," provides details or ground rules for complying with the LCOs.

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 analysis 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. 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, that since snubbers are supporting safety equipment that is in the TS, 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 TS 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 TS 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.

To remove the inconsistency in the treatment of snubbers among plants, 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,
- Avoiding reduced snubber testing and, thus, increasing the availability of snubbers to perform their supporting function,
- 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[1]

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:

---

[1] The following technical evaluation is presented in terms of the bounding assessment of this change for all commercial nuclear power plants, including IP2 and IP3, performed as part of the approval of TSTF-372, Revision 4, and publication of the CLIIP notices.

- 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 TS. 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-eighties 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 <u>all</u> 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 AFW. 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 AFW 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 U.S. 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 U.S. 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 U.S. 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 U.S. and West Cost 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 harder to analyze generically, than for seismic loads. For these reasons, licensees will be required to confirm every time LCO 3.0.8a is used, that at least one train of each system that is supported by the inoperable snubber(s) would remain capable of performing their 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 TS 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 AFW in a seismic event of magnitude up to the plant's safe-shutdown earthquake (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 loss-of-coolant accidents, 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 TS, 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 1 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 TS 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 2.

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 TS. 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 TS.

The NRC staff finds that the risk assessment results support the proposed addition of LCO 3.0.8 to the TS. 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 TS, 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 (HPCI) or reactor core isolation cooling (RCIC) 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 (LPCI) or core spray (CS)) 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 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.

- 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 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 TS 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, 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 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 (a)(4) itself.

3.2 Summary and Conclusions

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. Some potential undesirable consequences of this inconsistent treatment of snubbers are:

- Performance of testing during crowded windows when the supported system is inoperable with the potential to reduce the snubber testing to a minimum since the relocated snubber requirements 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.

- 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.

To remove the inconsistency among plants in the treatment of snubbers, licensees are 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. 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 (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 TS 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 TS 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 TS 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 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.

   (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.

   (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.

   (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. Should licensees implement 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 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 TS. 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.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York 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. 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 (70 FR 48203; August 16, 2005). 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

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

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Date: February 13, 2006