March 19, 2008

Mr. John T. Carlin
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: ADDITION OF LIMITING CONDITION FOR OPERATION 3.0.8 FOR INOPERABLE SNUBBERS (TAC NO. MD7037)

Dear Mr. Carlin:

The Commission has issued the enclosed Amendment No. 104 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated October 17, 2007, as supplemented by letter dated December 13, 2007.

The amendment modifies Technical Specifications requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.8. This operating license improvement was made available by the Nuclear Regulatory Commission on May 4, 2005 (70 FR 23252) as part of the consolidated line item improvement process

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

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 104 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

March 19, 2008

Mr. John T. Carlin
Vice President R. E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: ADDITION OF LIMITING CONDITION FOR OPERATION 3.0.8 FOR INOPERABLE SNUBBERS (TAC NO. MD7037)

Dear Mr. Carlin:

The Commission has issued the enclosed Amendment No. 104 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated October 17, 2007, as supplemented by letter dated December 13, 2007.

The amendment modifies Technical Specifications requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.8. This operating license improvement was made available by the Nuclear Regulatory Commission on May 4, 2005 (70 FR 23252) as part of the consolidated line item improvement process

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

Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 104 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

Package No.: ML080530056
Amendment No.: ML080530061
Tech Spec No.: ML080530065

OFFICE	LPL1-1/PM	LPL1-1/LA	ITSB/BC	LPL1-1/BC
NAME	DPickett	SLittle	GWaig	MKowal
DATE	3/12/ 08	2/28/ 08	2 / 21 / 08	3/13/ 08

OFFICIAL RECORD COPY

DATED: March 19, 2008

AMENDMENT NO. 104 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R.E. GINNA NUCLEAR POWER PLANT

PUBLIC
LPLI-1 R/F

M. Kowal	RidsNrrDorlLpla
S. Little	RidsNrrLASLittle
D. Pickett	RidsNrrPMDPickett
G. Hill (2)	
OGC	RidsOgcMailCenter
ACRS	RidsAcrsAcnwMailCenter
G. Dentel, RI	RidsRgn1MailCenter
G. Waig	RidsNrrDirsItsb
A. Lising	

cc: Plant Service list

cc:

Mr. Michael J. Wallace, President
Constellation Energy Nuclear Group, LLC
750 East Pratt Street, 18th Floor
Baltimore, MD 21202

Mr. John M. Heffley, Senior Vice President
and Chief Nuclear Officer
Constellation Energy Nuclear Generation
Group
111 Market Place
Baltimore, MD 21202

Kenneth Kolaczyk, Sr. Resident Inspector
R.E. Ginna Nuclear Power Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, NY 14519

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. John P. Spath
New York State Energy, Research,
and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Tonko
President and CEO
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Carey W. Fleming, Esquire
Senior Counsel - Nuclear Generation
Constellation Energy Nuclear Group, LLC
750 East Pratt Street, 17th Floor
Baltimore, MD 21202

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mr. George Bastedo, Director
Wayne County Emergency Management
Office
Wayne County Emergency Operations
Center
7336 Route 31
Lyons, NY 14489

Ms. Mary Louise Meisenzahl
Administrator, Monroe County
Office of Emergency Preparedness
1190 Scottsville Road, Suite 200
Rochester, NY 14624

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Brian R. Weaver
Director, Licensing
R.E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

Mr. Louis S. Larragoite
Manager – Nuclear Licensing
Constellation Energy Nuclear Generation
Group
111 Market Place, 2nd Floor
Baltimore, MD 21202

R.E. GINNA NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-244

R.E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 104
Renewed License No. DPR-18

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

   A. The application for amendment filed by the R.E. Ginna Nuclear Power Plant, LLC, (the licensee) dated October 17, 2007, as supplemented by letter dated December 13, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-18 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 104, 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/

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: March 19, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 104

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

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

Remove	Insert
3	3

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

Remove	Insert
3.0-1	3.0-1
3.0-2	3.0-2
3.0-3	3.0-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-18

R.E. GINNA NUCLEAR POWER PLANT, LLC

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By application dated October 17, 2007 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML072950225), as supplemented by letter dated December 13, 2007 (ML073540526), R. E. Ginna Nuclear Power Plant, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the R.E. Ginna Nuclear Power Plant.

The proposed change 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. The change is based on Technical Specification Task Force (TSTF) change traveler TSTF-372, Revision 4, which 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 was published in the *Federal Register* on May 4, 2005 (70 FR 23252).

The letter dated December 13, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration.

2.0 BACKGROUND

On April 23, 2004, the Nuclear Energy Institute 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 TS program. These initiatives are intended to maintain or improve safety through the incorporation of risk assessment and management techniques in the TSs, while reducing unnecessary burden and making TS requirements consistent with the NRC's other risk-informed regulatory requirements, in particular the Maintenance Rule.

The proposed change adds a new LCO to the TSs. LCO 3.0.8 allows licensees to delay declaring an LCO not met for equipment that is supported by snubbers unable to perform their associated support functions when the risk associated with the delay 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.

In adding the new LCO to Section 3.0 of the TSs, LCO 3.0.1 would be revised to state that "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8."

3.0 REGULATORY EVALUATION

In Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR), the NRC established its regulatory requirements related to the content of the 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 TSs. As stated in 10 CFR 50.36(d)(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 specifications, until the condition can be met." 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 snubbers 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 were 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 percent 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 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 the 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.

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.

4.0 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 NRC'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 NRC 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 percent) of low probability (5 percent) 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 percent 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 a magnitude higher than 0.1g, bounds a detailed analysis that 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 percent) 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 3.0.8.b 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-percent 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 3.0.8.b 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. 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 an 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 Central and Eastern U.S. 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 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 would 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, water-hammer loads, steam-hammer 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.a 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.

4.1 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 Sections 4.1.1 through 4.1.3.

4.1.1 <u>Risk Impact</u>

The bounding risk assessment approach, discussed in Section 4.0, 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 percent. 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, except for West Coast PWR plants, that all safety systems are unavailable to mitigate the accident. 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 a 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 (below). The first row lists the conditional risk increase, in terms of CDF, DR_{CDF}, caused by the out-of-service snubbers (as assumed in the bounding analysis). The second and third rows list the ICCDP and the ICLERP values, respectively. For the case where all inoperable snubbers are associated with only one train (or subsystem) of the supported safety systems, the ICCDP was obtained by multiplying the corresponding DR_{CDF} value by the time fraction of the proposed 72-hour delay to enter the actions for the supported equipment. For the case where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety system, the ICCDP was obtained by multiplying the corresponding DR_{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 because 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 TS, and (2) testing of snubbers is associated with higher risk impact than is 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 NRC'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 TSs 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 Trains	Single Train	Multiple Trains
DR_{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 3 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.

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., DR_{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. In RG 1.182, the NRC staff did not take a position on the value of 1E-3/year. Since the assessed conditional risk increase, DR_{CDF}, is significantly less than 1E-3/year, NUMARC states that 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 TSs.

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

DR_{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 than 1E-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, based on the results of configuration-specific risk assessments required by 10 CFR 50.65(a)(4) or by other TSs, 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 TS, such as reduced frequency of snubber testing, increased safety system unavailability, and the treatment of snubbers impacting multiple trains.

4.1.2 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 3.0.8.a 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 3.0.8.a applies. Therefore, potentially high-risk configurations can be avoided by ensuring that such a success path exists when LCO 3.0.8.a applies. Based on a review of the accident sequences that contribute to the risk increase associated with LCO 3.0.8.a, 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 3.0.8.a is used.

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

    1. 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

    2. 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 3.0.8.b applies), it was assumed in the bounding analysis (except for West Coast plants) that all safety systems are unavailable to mitigate the accident. 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 3.0.8.b (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 3.0.8.b 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 3.0.8.b 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 3.0.8.b 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.

4.1.3 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 10 CFR 50.65(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. Because 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, whether the process is invoked by a TS or by (a)(4) itself.

4.2 Summary and Conclusions

The option to relocate the snubbers to a licensee-controlled document, as part of the conversion to the 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, or

- 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 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. 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 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 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 NRC staff's approval and inherent in the implementation of TSTF-372, R.E. Ginna Nuclear Power Plant 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 3.0.8.a 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 the system's 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 versus non-seismic), the implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall all 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. Because 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 that utilized in this Safety Evaluation, shall be followed.

In its submittal, the licensee stated that it reviewed the NRC staff’s evaluation, as well as the information provided to support TSTF-372, and has concluded that the justifications presented in the TSTF proposal and NRC staff safety evaluation are applicable to R.E. Ginna Nuclear Power Plant, and justify this amendment. Based on its own review, the staff agrees. Therefore, incorporating the aforementioned changes into the R.E. Ginna Nuclear Power Plant TSs are acceptable.

5.0 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.

6.0 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 (72 FR 65371). 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.

7.0 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.

Principal Contributor: A. J. Lising

Date: March 19, 2008

8.0 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, July 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. Nuclear Energy Institute, 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.