October 3, 2007

Mr. Keith J. Polson
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P. O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 - AUTHORIZATION UNDER 10 CFR 50.55a(a)(3)(i) FOR MODIFICATION OF CORE SHROUD STABILIZER ASSEMBLIES (TAC NO. MD4417)

Dear Mr. Polson:

By letter dated February 12, 2007, as supplemented by letters dated March 8, 2007, May 3, 2007, and June 4, 2007, Nine Mile Point Nuclear Station, LLC requested authorization to modify the core shroud stabilizer assemblies (tie rods) for Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The NMP1 core shroud was repaired in 1995 by the installation of four tie rod assemblies, as a Nuclear Regulatory Commission (NRC) staff-authorized alternative to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i). However, recent industry experience identified the need to modify the core shroud repair, which Nine Mile Point Nuclear Station, LLC planned to do during refueling outage 19 (RFO19) in the spring of 2007. Because the proposed repair modification is not included under the ASME Code, Section XI definition for repair or replacement, the design details of the proposed core shroud repair modification were submitted to the NRC for review and authorization as an alternative repair, pursuant to 10 CFR 50.55a(a)(3)(i).

The NRC staff verbally authorized the repair modification in a conference call on March 26, 2007, and the core shroud tie rods were modified during RFO19 in the spring of 2007. The NRC staff was notified on April 9, 2007, that a post-modification inspection identified minor deviations from the requirements specified in the February 12, 2007, letter. In letters dated May 3, 2007, and June 4, 2007, Nine Mile Point Nuclear Station, LLC submitted its evaluation and technical justification of these deviations.

The results of the NRC staff's review and evaluation of the four afformentioned submittals is provided in the enclosed safety evaluation (SE). Based on the SE, the NRC staff concludes that the core shroud repair modification for NMP1 provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for NMP1. All other requirements of the ASME Code, Sections III and XI, for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

/RA/

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

Docket No. 50-220

Enclosure:
Safety Evaluation

cc w/encl: See next page

not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

/RA/

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

Docket No. 50-220

Enclosure:
Safety Evaluation

cc w/encl: See next page

DISTRIBUTION:

PUBLIC	RidsNrrLASLittle	RidsNrrDciCvib	RidsNrrAcrsAcnwMailCenter
LPLI-1	RidsNrrPMMDavid	RidsRgn1MailCenter	
RidsOGCRp			

ADAMS Accession Number: ML072600195 NRR-028

OFFICE	LPLI-1/PM	LPLI-1/LA	CVIB/BC	OGC	LPLI-1/BC
NAME	MDavid	SLittle*	MMitchell	EWilliamson	MKowal
DATE	9/19/07	09/19/07	9/19/07	9/2/07	9/3/07

OFFICIAL RECORD COPY
* Via e-mail

Nine Mile Point Nuclear Station

cc:

Mr. Michael J. Wallace
President
Nine Mile Point Nuclear Station, LLC
c/o Constellation Energy Group
750 East Pratt Street
Baltimore, MD 21202

Mr. Mike Heffley
Senior Vice President and Chief
Nuclear Officer
Constellation Generation Group
1997 Annapolis Exchange Parkway
Suite 500
Annapolis, MD 21401

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

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 126
Lycoming, NY 13093

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

Mr. Paul D. Eddy
Electric Division
NYS Department of Public Service
Agency Building 3
Empire State Plaza
Albany, NY 12223

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

Mark J. Wetterhahn, Esquire
Winston & Strawn
1700 K Street, NW
Washington, DC 20006

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, NY 13126

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

Mr. Michael Balboni
Deputy Secretary for Public Safety
State Capitol, Room 229
Albany, NY 12224

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR AUTHORIZATION FOR MODIFICATION OF CORE SHROUD TIE RODS

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the alternative request submitted by Nine Mile Point Nuclear Station, LLC (the licensee) in its letter dated February 12, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070600206), as supplemented by letters dated March 8, 2007 (ADAMS Accession No. ML070750104), May 3, 2007 (ADAMS Accession No. ML071410137), and June 4, 2007 (ADAMS Accession No. ML071630246), for authorization under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i) for modification of the core shroud stabilizer assemblies (tie rods) for Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The licensee proposed to implement a modification to each of the four tie rod assemblies during refueling outage 19 (RFO19) in the spring of 2007.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the "Code") Class 1, 2, and 3 components is to be performed in accordance with Section XI, of the ASME Code and applicable editions and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if a licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NMP1 licensee submitted the subject request for the NRC staff's review and approval to use as an alternative repair under the provisions of 10 CFR 50.55a(a)(3)(i).

3.0 TECHNICAL EVALUATION

3.1 ASME Code Requirement

The core shroud tie rod assemblies are not included under ASME Code, Section XI. However, the licensee's proposed alternative was designed to comply with the requirements of the ASME Code, Section III, Division 1, Subsection NG, 2001 Edition, and 2003 Addenda.

3.2 Licensee's Basis for Requesting Alternative

3.2.1 Background

Industry experience has shown that boiling-water reactor (BWR) core shroud welds are subject to intergranular stress-corrosion cracking and/or irradiation-assisted cracking (IGSCC/IASCC). As a result, the NRC staff issued Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds at Boiling Water Reactors," which led BWR facilities to perform detailed inspections and analysis of the core shrouds to determine susceptibility to this phenomenon. NMP1 installed core shroud tie rod assemblies in 1995 on a preemptive basis in lieu of ultrasonic (UT) inspection of the core shroud horizontal welds. The tie rods functionally replace the shroud horizontal welds H1 through H7. General Electric Company (GE) designed and installed the NMP1 tie rod assemblies and also provided core shroud repairs using tie rods to many other domestic BWR plants. By letters dated January 6, 1995, and January 23, 1995, Niagara Mohawk Power Corporation (NMPC), the former licensee, proposed a repair for the NMP1 core shroud by installing four tie rods to the core shroud. The NRC staff, in its letter dated March 31, 1995, found the proposed repair acceptable as allowed by the requirements of 10 CFR 50.55a(a)(3)(i).

In spring 2006, cracking was discovered in the tie rod upper supports at Hatch, Unit No. 1 during an in vessel visual inspection. The apparent root cause is IGSCC in the Alloy X-750 tie rod upper support material. Alloy X-750 material is susceptible to IGSCC if subjected to sustained large peak stress conditions. GE conducted an evaluation to determine if the potential IGSCC in the Alloy X-750 tie rod structural components of other BWR core shroud repairs designed by GE could be a reportable condition under 10 CFR Part 21. GE used the criterion provided in the BWR Vessel and Internals Project (BWRVIP) report BWRVIP-84, "BWR Vessel and Internals Project, Guidelines for Selection and Use of Materials for Repairs to BWR Internals," for the IGSCC susceptibility assessment of the Alloy X-750 components in the tie rod assembly. Based on the Hatch, Unit No. 1 finding, GE revised the assessment of the GE tie rod upper support design life and determined that the potential for a high peak surface stress existed for the NMP1 tie rod design. The potential for high peak stress in the original tie rod upper bracket design at NMP1 was attributed to the lack of a specified radius at the corner junction between horizontal and vertical legs of the bracket which creates a high stress concentration. This high peak stress reduced the design life of the original tie rod upper support. Tie rod inspections performed during each refueling outage could justify continued operation on a cycle-by-cycle basis, and such an inspection was being planned as a contingency for RFO19. However, the licensee determined that the most prudent course of action, and the best long-term economic solution, was preemptive replacement of the tie rod upper support with a modified upper support design capable of operation through the end of the renewed operating license term (year 2029).

GE conducted a review to determine if other Alloy X-750 tie rod components had similar potential for high peak stress. GE identified that the root radii of the threads in the tie rod threaded components of the original NMP1 core shroud repair may be smaller than the nominal values used in previous design evaluations. Hence, the licensee proposed to include a modified tie rod nut that incorporates an improved locking mechanism in the modified core shroud repair installed during RFO19. To improve IGSCC resistance, the new tie rod nuts include a specified root radius sufficient to minimize the peak principle stress to within the same criterion as used for the upper support. In the licensee's February 12, 2007, submittal, the licensee addressed the impact of the proposed modification on previously performed design analysis and evaluations. No other modifications to the existing tie rod assembly components were planned for RFO19.

3.3 Licensee's Proposed Alternative

3.3.1 Design Objectives

The objective of the proposed tie rod repair modifications is to design and install replacement upper support assemblies and tie rod top nuts that will remain resistant to IGSCC over the remaining plant life (year 2029).

3.3.2 Design Criteria

The modified upper supports and tie rod top nuts comply with the criteria delineated in report BWRVIP-02A, "BWR Vessel And Internals Project, BWR Core Shroud Repair Design Criteria," and the BWRVIP-84 report with no exceptions taken. The original codes and design standards used for construction of the original tie rod assemblies were delineated in GE Specification 25A5583, which was included in the 1995 core shroud repair submittals (References 1 and 2). The original codes and design standards remain applicable to the proposed modifications, as well as other more recent standards (e.g., BWRVIP-84), as discussed in later sections of this safety evaluation (SE).

3.3.3 Description of Repair Modification Components and Design Features

The geometry of the replacement hardware (upper support, tie rod nut, and other associated upper support components) is shown in Figure 1 of GE stress analysis report GE-NE-0000-0061-6180-R2-P, "Nine Mile Point 1 Nuclear Plant Shroud Repair Replacement Upper Support Stress Analysis Report," which was attached to the licensee's February 12, 2007, request. These newly-designed components incorporate features that improve their ability to resist IGSCC. These features include: (1) a large fillet radius at the corner of the upper support; (2) increased width and thickness of the upper support arms; (3) no sharp edges; and (4) a larger root radius of the tie rod nut threads. Additional details regarding the improvements made are provided in GE stress analysis report GE-NE-0000-0061-6180-R2-P. The original tie rod installation required that cutouts be made in the shroud head flange to accommodate the upper support arms which hang over the shroud flange. The width of the cutouts will be increased to accommodate the increased width of the modified upper supports arms.

The licensee submitted the original design analysis on January 6, 1995, which was approved by the NRC staff in a letter dated March 31, 1995. The licensee stated that the proposed modification has an insignificant affect on the following attributes of the original design analysis for all the items in the original tie rod assembly with the exception of the upper support assembly and the tie rod top nuts.

(1) Seismic model

(2) Structural model

(3) Original analysis on the cracked core shroud

(4) Load combinations

(5) Core shroud deflections

(6) Shroud deflections

(7) Effect of the tie rod modification on core shroud shell, shroud head and shroud support plate

(8) Effect of the tie rod modification on the reactor vessel and its internals

(9) Flow induced vibration

(10) Radiation effects

(11) Downcomer flow evaluation

3.3.4 Structural Evaluation and Design of the Core Shroud Repair Modifications

GE stress analysis report GE-NE-0000-0061-6180-R2-P, along with the alternative request dated February 12, 2007, addresses the stress analysis of the proposed tie rod modifications and defines loads that are applicable to these modifications. The applicable normal, upset, emergency and faulted loading combinations remain consistent with the original design basis of the shroud repair tie rods. The loads and load combinations are also in accordance with the BWRVIP-02A report, and the stress analysis report concludes that the replacement hardware is structurally qualified and is consistent with the original design specification as amended for improved IGSCC resistance. The replacement hardware is also structurally qualified in accordance with the ASME Code, Section III, Subsection NG allowable stress values including the fatigue evaluation. The details of the structural evaluation for the proposed tie rod modification and the NRC staff's evaluation is described in Section 4.0 of this SE.

The licensee's structural analysis of the internal components meet the primary and secondary stress requirements of ASME Code, Section III. The upper support and tie rod nut were structurally evaluated using the same load combinations as described in the BWRVIP-02A report. The modified upper support and tie rod nut stresses also were evaluated in accordance with ASME Code, Section III Subsection NG. The original tie rod components were evaluated using the COSMOS structural analysis program, which was approved by the NRC staff in its SE dated March 31, 1995. In the licensee's request dated February 12, 2007, GE used the ANSYS software program to perform a structural evaluation of the core shroud repair modification components. The ANSYS software program is a controlled safety-related program and GE has established that this software program as implemented for NMP1 is equivalent to the original design analysis with COSMOS software program. The NRC staff has accepted the use of the

ANSYS software program for the analysis of GE tie rod designs, most recently for the tie rod repair at the Clinton Nuclear Power Station, Unit No. 1. The licensee concluded that the ANSYS software program does not represent a different method of evaluation, and the original design analysis remains unchanged.

3.3.5 Bypass Flow

The proposed modification increased the width of the upper support arms and the upper support cutouts in the shroud head. The increase in the leakage due to this modification meets acceptance criterion specified in the original analysis. The acceptance criterion for increased leakage through the larger shroud head cutouts is that the combined bypass leakage of steam through the enlarged cutouts shall be less than the core flow minus steam flow for normal differential pressure. This criterion is specified in the original tie rod repair which was approved by the NRC staff in its SE dated March 31, 1995.

3.3.6 Downcomer Flow Characteristics

The licensee stated that the original tie rod design evaluation included an analysis of the available flow area in the downcomer region with the four tie rod assemblies installed. The original calculations showed that the downcomer flow area in the upper annulus region would be reduced with the tie rods installed. This resulted in an increase in upper annulus region flow velocity. The NRC staff concluded in its SE dated March 31, 1995, that the corresponding pressure drop is insignificant and would not affect the recirculation flow in the reactor vessel. The proposed modification with wider upper supports results in a small reduction in the total annulus flow area, which in turn increases the upper annulus region flow velocity. The original conclusions that the corresponding pressure drop is insignificant and that it would not affect the recirculation flow in the reactor vessel remain unchanged.

3.3.7 Materials and Fabrication

For the proposed modification, the licensee used the following materials:

(1) Alloy X-750—Tie rod upper supports that are located in main load path, and miscellaneous smaller parts that are not in main load path

(2) Alloy X-750—Tie rod nuts

(3) Type 316 stainless steel—Tie rod upper support dowel pins

The above-listed materials have been used for many other reactor vessel internals (RVI) components and have demonstrated good resistance to stress corrosion cracking in laboratory testing and long-term service experience in the non-welded and low sustained operating stress conditions. Both Alloy X-750 and Type 316 austenitic stainless steel are acceptable per the BWRVIP-84 report and Section III of the ASME Code. The proposed materials for the replacement parts are consistent with those used in the original NMP1 tie rod design, which was found to be acceptable as documented in the NRC staff's SE dated March 31, 1995.

Consistent with the fabrication requirements specified in the BWRVIP-84 report, the licensee proposed to not utilize any avoidable crevices in the upper bracket design. If the crevices are inherently present, the licensee proposed to implement the following requirements:

(1) The design of crevice will not have any stainless steel material with sensitized microstructure, and only IGSCC resistant materials will be used, and

(2) To the extent practical, stagnant conditions will be minimized.

To avoid IGSCC, consistent with the requirements of the BWRVIP-84 report, the licensee used only non-sensitized stainless steel materials without any welds in the replacement upper support assemblies.

3.3.8 Pre-Modification Inspection

According to the licensee, the pre-modification inspection included a video recording of the as-found condition of the shroud repair tie rod assemblies. This inspection was intended to confirm that the tie rod assembly was properly installed and that the tie rod tightness requirements, per report BWRVIP-76,"BWR Core Shroud Inspection and Flaw Evaluation Guidelines," were met. The licensee performed inspection of the existing tie rod upper supports when access was provided during the planned replacement activity. The upper support inspection was performed utilizing an enhanced visual testing (EVT-1) examination of the high stress locations identified in the GE Part 21 notification dated October 9, 2006 (Reference 3).

In addition, an EVT-1 examination of the upper and lower tie rod and tie rod nut threads was performed to the extent accessible. The licensee also reviewed all of the Alloy X-750 tie rod assembly components located in the primary vertical and horizontal load paths. Based on this review, inspection of other similar high stress Alloy X-750 locations was performed consistent with the BWRVIP recommendations (References 4 and 5).

3.3.9 Radiation Effects

Neutron fluence estimates for the core shroud tie rods that were developed for the NMP1 license renewal application indicate that the maximum neutron fluence values at the end of the renewed operating license in year 2029 remain below the threshold limits for irradiation-assisted stress corrosion cracking. Radiation effects as they relate to design controls and material selection for the new upper supports and tie rod nuts are the same as those considered for the original parts. Since the projected neutron fluence values do not exceed the threshold limits, the licensee concluded that the mechanical properties of the core shroud tie rods are not affected.

3.3.10 Post-Modification Inspection

The licensee stated that it would perform a post-modification inspection prior to reactor vessel reassembly. This inspection would include a general post-maintenance visual inspection and recording of the fit-up of the shroud support hardware onto the shroud to confirm that there are no interferences at the support locations. This inspection would also verify that the installation was in accordance with the requirements of the GE's modification drawings and the installation specification.

3.3.11 Inspection During Subsequent Refueling Outages

In the first refueling outage following installation of the modified tie rod upper supports, the licensee plans to inspect the tie rod assemblies in accordance with the requirements defined in the BWRVIP-76 report, Section 3.5, Option 1 or 2. The licensee will repeat the post-installation inspections described in Section 7.2.1 of the licensee's submittal dated January 6, 1995.

3.2.12 Loose Parts and Installation Cleanliness

In its submittal dated February 12, 2007, the licensee indicated that the mechanism for capturing loose parts is designed for the design life of the repair modification and, therefore, loose parts generated in the proposed modification are adequately controlled. Electric discharge machining (EDM) that will be used for this modification is designed with a filtration system which is capable of filtering 99% of debris of a 2 micron in size. The licensee concluded that there will not be any debris in the reactor vessel and, therefore, plant components and fuel are not affected.

3.2.13 Conclusion

Based on the technical justification stated above, the licensee concluded that the proposed modification provides an acceptable level of quality and safety pursuant to the requirements specified in 10 CFR 50.55a(a)(3)(i).

4.0 NRC STAFF EVALUATION

By letters dated January 6, 1995, and January 23, 1995, NMPC, the former licensee, submitted a proposed repair of the NMP1 core shroud consisting of installation of four tie rods to the NRC staff for review and approval. The NRC staff, in its letter dated March 31, 1995, found the proposed repair acceptable per the requirements of 10 CFR 50.55a(a)(3)(i). After the discovery of IGSCC cracking of the tie rod upper support at Hatch, Unit No. 1, the NMP1 licensee voluntarily planned to replace the tie rod upper supports at NMP1 with a modified upper support design, which is addressed in the submittal dated February 12, 2007. The modified design has a large fillet radius at the corner of the upper support, which reduces the peak stress with an intent to increase the resistance to IGSCC. In addition, this modification includes a wider and thicker upper support without sharp edges and a larger root radius of the tie rod nut threads to improve resistance to IGSCC. The NRC staff reviewed the following items that are addressed in the licensee's submittal dated February 12, 2007, as supplemented by letters dated March 8, 2007, May 3, 2007, and June 4, 2007. The NRC staff's evaluation is discussed below.

4.1 Structural and Design Evaluation

The licensee stated that the proposed modification has an insignificant affect on the original design attributes as addressed in Section 3.3.3 of this SE, for all the items of the original tie rod assembly with the exception of upper support assembly and tie rod top nuts. The NRC staff in its SE dated March 31, 1995, approved the original design attributes associated with all the items of the original tie rod assembly. The proposed modification does not affect the original design of the items other than the upper support assembly and tie rod top nuts. Therefore, the NRC staff concludes that its original approval of the design for the items other than the upper support assembly and tie rod top nuts is still valid, and is acceptable. In its design review of the current tie rod modification, the NRC staff compared the load cases and load combinations used in the original analysis described in the licensee's submittal dated January 6, 1995, to the modification proposed in the licensee's submittal dated February 12, 2007, and its supplements.

The original analysis (dated January 6, 1995) included loading conditions for normal operation; two upset conditions; three faulted conditions, and three emergency conditions. However, the licensee, in the submittal dated February 12, 2007, used load cases representing one emergency condition and one faulted condition for the proposed modification. In a request for additional information (RAI), dated March 7, 2007, the NRC staff requested that the licensee provide an explanation for using load cases that are not consistent with the original design and confirm if the load cases presented for the proposed modification are bounding. The licensee, in its response dated March 8, 2007, stated that the stress analysis for the replacement upper support was performed using the bounding (largest) loads under the faulted and emergency conditions, and this analysis is consistent with that of the original design analysis. The NRC staff finds this response acceptable because the licensee used bounding analysis and this methodology is consistent with the NRC staff approved original design analysis. Further review of the design analysis indicated that the licensee used the following aspects of design criteria for the proposed modification of the upper shroud support and tie rod nuts.

(1) Maximum tensile principal stress due to sustained normal loading conditions for IGSCC evaluation of Alloy X-750 materials

(2) Stress intensities for the upper shroud support to demonstrate ASME Code compliance

In the proposed modification, the licensee used a criterion for IGSCC susceptibility of Alloy X-750 materials, which specifies a maximum threshold principal stress limit that is more conservative than the existing value specified in the BWRVIP-84 report. By designing an upper core shroud support fillet radius, the maximum principal stress is reduced significantly in the upper support, which reduces the component's susceptibility to IGSCC. The NRC staff reviewed the maximum principal stress values of all the Alloy X-750 materials that are used in the proposed modification and concludes that these principal stress values comply with the IGSCC design criterion proposed by the licensee. Compliance with this conservative IGSCC design criterion provides additional safety margin for all the Alloy X-750 materials used in the core shroud tie rod modifications.

The licensee used the ASME Code design criteria, as specified in ASME Code, Section III, Subsection NG, Core Support Structure, 2001 Edition through and including 2003 Addenda, for the core shroud tie rod modifications. All the stress intensity values of the components that are used in the core shroud tie rod modifications comply with the ASME Code, Section III, Subsection NG design criteria. Therefore, the NRC staff concludes that the licensee's core shroud tie rod modifications are structurally qualified to meet the ASME Code, Section III design criteria, and this design is consistent with the design basis of the original tie rod repair. To perform the design analysis, GE used ANSYS, which is a controlled finite element modeling program equivalent to the COSMOS software program which was used in the design analysis of the original core shroud tie rod repair. By a letter dated January 17, 2006, (ADAMS Accession No. ML060110504) the NRC staff accepted GE's use of the ANSYS software program for the core shroud tie rod repair modification for the Clinton Nuclear Power Station, Unit No. 1. Therefore, the NRC staff concludes that the application of the GE ANSYS software program for the design analysis of the NMP1 core shroud tie rod modifications is acceptable.

The fatigue evaluation of the original core shroud tie rod repair revealed that the licensee complied with the ASME Code, Section III, Subsection NG requirements. In addition, the fatigue usage factors for all the components that are used in the core shroud tie rod modification remain low, thus providing extra safety margin for remaining life of the plant.

With respect to the issues related to materials and their fabrication, the licensee used the guidelines specified in the BWRVIP-84 report which, in a letter dated September 6, 2005, was approved by the NRC staff with some conditions. The NRC staff, in a RAI dated March 7, 2007, requested that the licensee comply with the following conditions stipulated in the NRC staff's September 6, 2005, SE for the BWRVIP-84 report.

(1) Section 3.5.2 of the NRC staff's SE - Surface preparation of cold worked austenitic stainless steel materials:

Previous experience indicates that cold working of austenitic stainless steel RVI components increases susceptibility to IGSCC. Removal of cold worked area by adapting surface finishing techniques described in the BWRVIP-84 report is essential in reducing the susceptibility of austenitic stainless steel RVI components to IGSCC.

(2) Section 3.6.2 of the NRC staff's SE - Surface preparation of cold worked Alloy X-750 materials:

Previous experience indicates that cold working of Alloy X-750 RVI components increases susceptibility to IGSCC. Removal of cold worked area by adapting surface finishing techniques described in the BWRVIP-84 report is essential in reducing the susceptibility of Alloy X-750 RVI components to IGSCC.

(3) Section 3.6.3 of the NRC staff's SE - Surface preparation of EDM of Alloy X-750 materials:

Susceptibility to IGSCC is increased when EDM-cut surface of Alloy X-750 RVI components is exposed to reactor coolant water. Therefore, a minimum of 1/16" of material, or an amount demonstrated by the vendor to be adequate to assure that the surface is free of cracks and the resulting surface is acceptable for BWR service, shall be mechanically removed from surfaces processed by EDM. One approach for determining the amount of material that must be removed for the application is by use of the criteria contained in ASME Code, Section XI, IWA-4461.4. NRC staff approval is required if EDM-processed Alloy X-750 is used in a repair/replacement to a component which is subject to the requirements of ASME Code, Section XI.

The licensee in its response dated March 8, 2007, stated that it would comply with the NRC staff's aforementioned conditions for the core shroud tie rod modification. The NRC staff finds this response acceptable because the licensee's compliance with these conditions is consistent with the BWRVIP-84 guidelines, and implementation of these conditions ensures adequate mitigation of IGSCC.

4.2 Radiation Effects

Neutron fluence estimates that were developed in the NMP1 license renewal application for the core shroud tie rods indicate that the maximum neutron fluence values at the end of the renewed operating license in year 2029 remain below the IASCC threshold limits of 5×10^{20} n/cm^2 (energy greater than 1 MeV). These threshold limits are specified in NUREG-1801, "Generic Aging Lessons Learned," which is used for the evaluation of the license renewal applications. In June 2006, the NRC staff issued the final SE Report for the license renewal application, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Unit Nos. 1 and 2," (ADAMS Accession N0. ML061460313) in which the NRC staff accepted the licensee's evaluation because the projected neutron fluence on the core shroud components is below threshold limits for occurrence of IASCC. Therefore, the NRC staff concludes that the core shroud tie rod modification is bounded by this evaluation of the effects of radiation for the license renewal period.

4.3 Bypass Flow and Downcomer Flow Characteristics

The NRC staff accepts the licensee's disposition related to the bypass flow and downcomer flow characteristics because the original design criteria for these flow characteristics are not affected by the core shroud tie rod modification.

4.4 Loose Parts and Installation Cleanliness

After the review of the licensee's evaluation related to this issue, the NRC staff concludes that the licensee had proper measures in place to minimize the in-vessel debris generation and that it identified and evaluated any effects of the debris that remained in the reactor vessel after the repair modification is completed. In addition, the NRC staff noted that the BWRVIP-02A and BWRVIP-84 reports prohibit the use of cleansers or detergents that could introduce halides or sulfides, which could induce stress corrosion cracking of the internals. Since the licensee implemented cleaning and cleanliness control guidelines in accordance with the NRC staff-approved BWRVIP-02A and BWRVIP-84 reports, the NRC staff finds the licensee's cleanliness control methods acceptable.

4.5 Pre-Modification Inspections

Consistent with the requirements of the BWRVIP-76 report, the licensee performed pre-modification inspections of the shroud tie rod assemblies in the as-found condition. In addition, the licensee performed inspections of the high stress locations in the upper support as recommended after the Hatch, Unit No. 1 experience. Based on the inspection results and the recommendations specified in the BWRVIP-76 report, the licensee performed pre-modification inspection of high stress locations of Alloy X-750 components. The NRC staff finds the licensee's pre-modification inspections acceptable as these inspections provided valuable information regarding the structural integrity of the tie rod assemblies.

4.6 Post-Modification Inspections

The licensee's post-modification inspections prior to reactor vessel reassembly provided information regarding the proper installation fit-up of the shroud support hardware onto the shroud. Proper fit-up and installation of the tie rod assembly was essential in ensuring that the tie rods will perform their intended function during all design basis conditions. In a conference call with the NRC staff on April 9, 2007, the licensee identified minor deviations from the requirements specified items (e) and (g) in the post-modification section of its submittal dated February 12, 2007. In letters dated May 3, 2007, and June 4, 2007, the licensee submitted its evaluation of these deviations, which is discussed below.

In the first deviation (item e), contact is not established on both sides of the hook between the lower support clevis pin and the hook at the 166^0 and 270^0 locations resulting in reduction of tie rod thermal preload. The licensee's evaluation included the effect of a reduction in thermal preload on the operability and functionality of the tie rods with respect to the design criteria. Reduction in thermal preload reduces the clamping force, which may potentially cause separation of the limiting H6B weld under the limiting upset # 2 load conditions (upset pressure + dead weight + seismic). The NRC staff, in a telephone conference call on May 18, 2007, requested that the licensee provide an explanation as to how the reduction in clamping force on H6B weld can effectively restrain the lateral movement of the shroud assembly during the normal and/or upset conditions. In a letter response to the NRC staff's request dated June 4, 2007, the licensee indicated that there is still a positive net compressive force maintained on the shroud assembly even with the reduction in clamping force under normal loading conditions. For the upset # 2 load condition, the potential for lateral displacement of the core shroud as a result of reduction in the tie rod thermal preload is minimized by the upper and lower springs. The licensee evaluated the potential impact of the weld separation due to the reduction in thermal preload on the stresses of the upper spring and the lower spring of the tie rod assembly, and concluded that the stresses in these areas are lower than the maximum allowable values per the ASME Code, Section III, Subsection NG. Similarly, compliance with the ASME Code, Section III stress criteria ensures that the upper spring and the lower spring will not suffer any structural degradation as a result of reduction in thermal preload. Since the stress values of all the tie rod lateral support components remain below the maximum allowable ASME Code, Section III stress limits, the NRC staff accepts the licensee's aforementioned evaluation.

The licensee further evaluated the effect of a reduction of thermal preload on the following items and concluded that the original design margins are not affected as a result of reduction in thermal preload.

(1) Flow induced vibration

(2) Load carrying capability under loss-of-coolant accident conditions

The NRC staff agrees that the thermal preload reduction will not effect these issues for the core shroud repair modification. Since the reduction in thermal preload does not affect the original design margins for aforementioned items, the NRC staff concludes that the subject deviation left uncorrected will not compromise the safety function of the tie rod modification.

With regards to the stresses in the shroud shell, the licensee claimed that under normal operation there is sufficient compression in the shroud shell welds to prevent any core shroud weld separation. The NRC staff agrees with this disposition because the subject uncorrected deviation would not eliminate the required compression that is necessary to prevent any weld separation.

In response to the NRC staff's request related to the uneven thermal preload distribution at tie rod locations other than ones at the 166^0 and 270^0 locations, the licensee, in a letter dated June 4, 2007, stated that under normal operating conditions the shroud repair as a whole will adequately maintain compression, and the load distribution is independent of thermal preload. Therefore, the shroud repair modification will adequately maintain its function. Under the upset # 2 load conditions, the uneven preload does not affect the function of the tie rod modification because direct tension in the tie rods would resist the vertical load. Since the load distribution is not affected by reduction in thermal preload, the NRC staff finds the licensee's evaluation acceptable.

In the second deviation (item g), the lower wedge is not flush with lower spring. The licensee evaluated this deviation taking into consideration its effect on compliance with the design requirements for IGSCC resistance specified in the BWRVIP-84 report for Alloy X-750 retainer clip. The licensee concluded that the peak principle stress in the retainer clip is less than the design criteria for IGSCC resistance specified in the BWRVIP-84 report even when the lower wedge is left uncorrected. In addition, the lower wedge and the retainer clip will be inspected during each refueling outage and the position of lower wedge will be adjusted to ensure that the stresses in the retainer clip are bounded by the original design criteria specified in the BWRVIP-84 report. The NRC staff accepts the licensee's conclusion based on the fact that the principle stress values of the Alloy X-750 retainer clip will comply with the design criteria specified in the BWRVIP-84 report. Additional verification of this design compliance through future inspections of the retainer clip per the BWRVIP-76 guidelines provides reasonable assurance that the tie rod modification will continue to function satisfactorily with the subject deviation left uncorrected.

The licensee made a commitment to perform an inspection after one cycle of operation to verify that the clevis pin to the lower support hook interface remains consistent with the spring 2007 RFO19 as left configuration. The NRC staff concludes that the licensee's commitment is acceptable providing the licensee takes appropriate corrective actions if the inspection results indicate a change in tie rod configuration. In addition, the NRC staff accepts the licensee's commitment to perform inspections of the tie rod assembly per the NRC staff-approved BWRVIP-76 report during subsequent refueling outages. Implementation of the inspection guidelines of the BWRVIP-76 report provide adequate information regarding the structural integrity of the core shroud repair modification.

Based on the aforementioned technical bases, the NRC staff concludes that the tie rod modifications are acceptable with noted deviations left uncorrected.

The NRC staff finds that the core shroud repair modification for NMP1 is acceptable for the following reasons:

(1) The newly-designed tie rod upper support brackets have wider and thicker upper arms with no sharp edges, and have a larger root radius of the tie rod nut threads. A large fillet radius at the corner of the upper support arms reduces the peak stress and increases resistance to IGSCC.

(2) The licensee’s core shroud tie rod repair modifications are structurally qualified to meet the ASME Code, Section III design criteria, and this design is consistent with the original design basis for the tie rod repair.

5.0 CONCLUSIONS

Based on the above discussion, the NRC staff concludes that the core shroud repair modification implemented at NMP1 provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for NMP1. All other requirements of the ASME Code, Sections III and XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

6.0 REFERENCES

(1) Letter from C. D. Terry (NMPC) to Document Control Center Desk----NRC, dated January 6, 1995, Generic Letter 94-03, “Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors.” (TAC No. M90102)

(2) Letter from C. D. Terry (NMPC) to Document Control Center Desk----NRC, dated January 23, 1995, Generic Letter 94-03, “Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors.” (TAC No. M90102)

(3) Letter from J. S. Post (GE) to Document Control Center Desk----NRC, dated October 9, 2006, Part 21 Notification: Completion of GE Evaluation on Core Shroud Repair Tie Rod Upper Support Cracking.

(4) Letter from W. Eaton (BWRVIP) to all BWRVIP Committee Members, dated March 29, 2006, BWRVIP Recommendation to Inspect Core Shroud Tie Rod Repairs.

(5) Letter from R. Dyle/T. Mulford (BWRVIP) to all BWRVIP Committee Members, dated April 3, 2006, Clarification to BWRVIP Recommendation to Inspect Core Shroud Tie Rod Repairs.

Principal Contributor: Ganesh S. Cheruvenki

Date: October 3, 2007