UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MEMORANDUM TO: Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects I/II

FROM: Richard H. Wessman, Chief
Mechanical Engineering Branch
Division of Engineering

SUBJECT: SAFETY EVALUATION OF THE REPAIR PROPOSAL FOR THE NINE MILE POINT 1 CORE SHROUD - NIAGARA MOHAWK POWER CORPORATION

By letter dated January 6, 1995, as supplemented by letters dated January 23, and 26, February 14, 24, and 28, March 7 and 9, two on March 13, two on March 14, March 23, 27 and 28, 1995, Niagara Mohawk Power Corporation (NMPC) submitted the details of the planned repair of the circumferential welds for the Nine Mile Point Unit 1 (NMP1) reactor core shroud. Information was also provided to the staff during conference calls held on March 1, 3, 23, 24, and 27, 1995. Initially, the licensee's planned repair involved installation of four tie-rod assemblies combined with core plate wedges to replace welds H1 through H7, and six brackets to replace the downward vertical load capability of the H8 weld. It was NMPC's intention to examine the H1 through H8 shroud welds in accordance with the BWRVIP Inspection Criteria and install the tie-rod assemblies and/or the H8 weld brackets only if cracking was found to be unacceptable for continued plant operation. Based on the results of the ultrasonic examination of the H8 weld, NMPC decided to install the four tie-rod assemblies and not the brackets.

As discussed in the attached Safety Evaluation, based on a review of the shroud modification hardware from structural, systems, materials and fabrication considerations, the staff finds the proposed modifications of the NMP1 core shroud acceptable.

Attachment: Safety Evaluation

Docket No.: 50-220

Contact: J. Rajan, NRR
415-2788

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SAFETY EVALUATION OF THE REPAIR PROPOSAL
FOR THE NINE MILE POINT UNIT 1 CORE SHROUD
NIAGARA MOHAWK POWER CORPORATION
DOCKET NUMBER 50-220

1.0 BACKGROUND

In Boiling Water Reactors (BWRs) the core shroud is a stainless steel cylinder within the reactor pressure vessel (RPV) that provides lateral support to the fuel assemblies. The core shroud also serves to partition feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core.

In 1991 cracking of the core shroud was visually observed in a foreign BWR. The crack in this BWR was located in the heat affected zone of a circumferential weld in the mid shroud shell. The General Electric Company (GE) reported the cracking found in the foreign reactor in a Rapid Information Communication Services Information Letter (RICSIL) 054. GE identified the cracking mechanism as intergranular stress corrosion cracking (IGSCC).

A number of domestic BWR licensees have recently performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic experience. The cracking reported in the Brunswick Unit 1 core shroud was particularly unique since it was the first time that extensive 360° shroud cracking had been reported by a licensee in a domestic BWR. The 360° shroud crack at Brunswick Unit 1 was located at weld H3 which joins the top guide support ring to the mid shroud shell. Information Notice

93-79 was issued by the NRC on September 30, 1993, in response to the observed cracking at Brunswick Unit 1.

The cracks reported by the Commonwealth Edison Company (ComEd, the licensee for the Dresden, Lasalle, and Quad Cities units) in the Dresden Unit 3 and Quad Cities Unit 1 core shrouds were of major importance, since they signified the first reports of 360° cracking located in lower portions of BWR core shrouds. These 360° cracks are located at shroud welds H5, which join the core support plate rings to the middle shroud shells in the Dresden and Quad Cities Units. Information Notice 94-42 and its Supplement were issued by the NRC on June 7 and July 19, 1994, respectively, to alert other licensees of the shroud cracking discovered at Dresden Unit 3 and at Quad Cities Unit 1.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03 to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in the reactors' core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to the core shrouds:

- inspect the core shrouds no later than the next scheduled refueling outage;

- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;

- develop an inspection plan which addresses inspections of all shroud welds, and which delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject;

- develop plans for evaluation and/or repair of the core shroud;

- work closely with the BWROG on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking.

By letter dated January 6, 1995, as supplemented by letters dated January 23, and 26, February 14, 24, and 28, March 7 and 9, two on March 13, two on March 14, March 23, 27 and 28, 1995, Niagara Mohawk Power Corporation (NMPC) submitted the details of the planned repair of the circumferential welds for the Nine Mile Point Unit 1 (NMP1) reactor core shroud. Information was also provided to the staff during conference calls held on March 1, 3, 23, 24, and 27, 1995. Initially, NMPC's planned permanent repair involved installation of four tie-rod assemblies combined with core plate wedges to replace welds H1 through H7 and six brackets to replace the downward vertical load capability of the H8 weld. It was NMPC's intention to examine the H1 through H8 shroud welds in accordance with the BWRVIP Inspection Criteria and install the tie-rod assemblies and/or the H8 weld brackets only if cracking was found to be unacceptable for continued plant operation. Based on the results of the ultrasonic examination of the H8 weld, NMPC decided to install the four tie-rod assemblies and not the brackets.

2.0 EVALUATION

2.1 Scope of the Modification Design

The licensee indicated that the design life of all repair hardware is twenty-five years (the remaining life of the plant, plus life extension beyond the current operating license) which accounts for twenty effective full power years. The proposed modification takes into account 3, 4, or 5 recirculation pump operation, 105% core flow, and fluctuations in feedwater temperature during normal operations including loss of feedwater heating with a scram. The proposed modification is intended to maintain the structural integrity of the shroud with postulated 360° throughwall failure of welds H1 through H7. Thus, the functions of these welds is replaced with four stabilizer assemblies. The NMP1 repair of the core shroud is considered a non-ASME code repair and therefore is performed as an alternative to the ASME Section XI pursuant to 10 CFR 50.55a(a)(3).

2.2 Shroud Stabilizer Design Description

The design of the NMP1 core shroud modification consists of four sets of stabilizer assemblies, which was installed approximately 90° apart. Each stabilizer assembly consists of an upper spring, an upper bracket and tie rod support, a tie rod, a mid-span tie rod support, a lower spring, a lower anchor assembly, and other minor parts. The tie rod provides the vertical load carrying capability from the upper bracket to the lower anchor assembly attached to the RPV core shroud support cone, and provides support for the springs. The vertical locations of the radial springs were chosen to provide the maximum support for the shroud, top guide, core plate, and, the fuel assemblies. The upper spring provides radial load carrying capability from the shroud, at the top guide elevation, to the RPV. The lower spring provides radial load carrying capability from the shroud, at the core support plate elevation, to the RPV. The upper stabilizer bracket provides an attachment feature to the top of the shroud as well as restraint of the upper shroud welds. The mid-span tie rod support is installed to provide a limit stop for the shroud cylinder between the H4 and H5. The mid-span tie rod support which is preloaded against the RPV effectively divides the tie rod into two shorter, stiffer rods to increase the natural frequency of the tie rod assembly, thereby preventing unacceptable levels of flow-induced vibration. At the top, each stabilizer assembly fits through two slots, which are machined into the non-safety-related shroud head and steam surface of the shroud top flange. The assembly then extends downward to below weld H3. The stabilizer assembly supports the upper spring and has a hole through which the tie rod passes. The tie rod is held against the upper bracket with a nut. The tie rod extends downward approximately 136 inches and is threaded into the lower spring. The lower spring has a pin at the bottom, which is attached to the clevis in the lower support. The lower support is bolted to the shroud support cone with two toggle bolts. The primary forces that the stabilizers would experience are from seismic events, LOCA differential pressure loads, and differential thermal expansion. The stabilizer assemblies and cracks in the shroud change the seismic response of the reactor internals. Thus, it was necessary to modify the seismic analysis of the reactor to include the effects of the cracks and the stabilizers. This dynamic analysis was performed in an iterative manner to determine the appropriate values of the spring constants of the upper and lower springs as well as the number of stabilizer assemblies required. The analysis results indicated that four stabilizer assemblies would be acceptable.

2.3 STRUCTURAL EVALUATION

2.3.1 Stabilizer Assemblies

The stabilizers were designed to the structural criteria specified in the NMP1 UFSAR. The UFSAR compares the calculated shroud stresses against the allowable stress (Sm) for all operating conditions and events. Allowable stress intensities for other stress combinations and accident conditions are not addressed in the UFSAR. The purchase specification for the RPV designates the following allowable stress limits. The primary membrane stress is limited to Sm, 1.5 Sm and 2.0 Sm during normal/upset, emergency and faulted events, respectively. The primary membrane plus bending stress is limited to 1.5 Sm, 2.25 Sm and 3 Sm during normal/upset, emergency and faulted events, respectively. The shear stress is limited to 0.6 Sm, 0.9 Sm and 1.2 Sm during normal/upset, emergency and faulted events, respectively. These allowable stress intensities are consistent with the allowables used in other shroud designs reviewed by the staff. The staff finds these allowable stresses acceptable. All of the loads and load combinations specified in the UFSAR, that are relevant to the core shroud, were evaluated in the design. The stabilizers are installed with a small tension preload of 3,000 lbs., to ensure that all components are tight. The stabilizer assemblies will be thermally preloaded to 79,670 lbs. during normal operating conditions. This tensile load in the tie rod results from the thermal expansion coefficient for the new stabilizer hardware being less than the thermal expansion coefficient of the shroud. The maximum permanent horizontal deflection of any part of the shroud that is not directly supported by either the upper or lower radial springs is limited to approximately 0.75 inches by mechanical limit stops. These stops do not perform this function unless a section of the shroud, for example between H4 and H5, becomes loose and a combined LOCA plus seismic event occurs. If this scenario occurs, the stops will limit the horizontal displacement to approximately 0.75 inches which is equal to one-half of the shroud wall thickness. A displacement equal to one-half of the shroud wall thickness will not result in post event leakages that prevent core cooling, because the shroud sections still overlap each other by one-half (0.75 inches) of the shroud wall thickness.

Wedges between the core support and the shroud (also called the Clamp/Spacer) are required at each stabilizer location to prevent relative motion of the core plate to the shroud. The four spacers are located in the annulus between the core support and the shroud and rest on the shroud ring. The wedges are held in place by clamping under the existing angle brackets that position the existing shield blocks. The annulus is measured at each location and the spacers are machined for a maximum clearance of 0.030 inches at the core plate elevation. In the event that welds H6A and H6B should fail, the wedges would provide a direct load path from the core plate to the shroud to help distribute the lateral loads occurring during a seismic event. The shroud cylinder at this location is restrained in the lateral direction by the lower tie rod spring.

The upper and lower springs of the stabilizers are installed with a small radial preload such that they provide radial support for the shroud. During normal operation, the shroud and stabilizer springs radially expand due to thermal growth slightly more than the RPV, which increases the radial preload and assures that the springs provide lateral support for the shroud during normal operation.

The vertical locations of the upper and lower springs were chosen to provide the maximum horizontal support for the fuel assemblies. The upper springs are at the top guide elevation and the lower springs are at the core support plate elevation. All of the horizontal support for the fuel assemblies is provided by the top guide and the core support plate.

A detailed finite element model, using the COSMOS code of the NMP1 shroud and repair assembly, was developed for stress analysis purposes to fully evaluate all of the loading conditions specified in GE Design Specification No. 25A5583, Rev. A, "Shroud Repair Hardware." The model consisted of a 180° shroud segment that incorporated the shroud shell, gaps (representing cracks), vertical tie rod assemlies/repair springs, and lower brackets. Repair spring and vertical tie rod assemblies were included in the model as 3-D truss elements and lower brackets as 3-D beam elements representing the repair hardware global mechanical characteristics. A 180° segment was necessitated by the need to evaluate the non-symmetric loads.

The shroud spring and vertical tie rod components were separately modeled in detail to evaluate their mechanical characteristics and behavior. These models are described in detail in the licensee submittal GE-NE-B13-01739-04, Rev. B.

The COSMOS finite element code has been verified for use in the nuclear power industry in accordance with the requirements of 10 CFR 50, Appendix B and the applicable sections of ANSI/ASME QA-1 and related supplements.

According to the licensee, the COSMOS code users' guide documents a close comparison between finite element analysis results and closed form solutions for over 1000 problems of different type elements and loading conditions. For validating the COSMOS code for NMP1 application, the verification problems for the elements used in the shroud analysis (Solids, 3-D beam, rigid bar, spring, coupling and gap) were reanalyzed by the licensee. The results of the reanalysis, according to the licensee, are in good comparison with the closed form solutions.

Based on its review of the analysis presented by the licensee in its submittal GE-NE-B13-01739-04, "Shroud Repair Hardware Stress Analysis - Nine Mile Point Unit 1" and related documents, the staff finds that the maximum stresses in the tie-rods, upper and lower springs and supports including the shroud conical support remain within the allowables for applicable normal, upset and faulted conditions. Therefore, the structural integrity of the shroud and repair hardware is maintained after the proposed repairs. However, if an upset condition occurs, the licensee should evaluate the effect of the event on the shroud and the tie rod assemblies (including the preload) prior to returning to power operation.

2.3.2 Evaluation of Postulated Critical Weld Failures

The licensee analyzed the worst-case scenario for 360° through-wall cracking in all the circumferential welds from H1 through H7. Since cracking at welds H2 and H3 could affect the shroud stiffness, and therefore the preload, additional stress analysis was performed to evaluate this condition. The results confirm that gaps would not develop under normal operating conditions for cracks at welds H1 through H7. For upset conditions, conservative assumptions predict a maximum separation of .030 inches. The existence of gaps during conditions other than normal operation does not violate the generic VIP shroud repair guidelines. The potential crack separation for upset event conditions is temporary and is projected to close following the event since the thermal preload will be recovered. The licensee's calculations indicate that the installation pre-load would not be affected following an upset event and that the calculated tie rod assembly stresses would remain within elastic limits. Realistic assumptions regarding the H2 and H3 fillet weld integrity demonstrate that no separation would occur for bounding 100% rated core flow upset condition pressures. In the evaluation for faulted accident conditions, gaps are predicted at several weld locations. An assessment of the conseqences from this event was provided in a previous safety evaluation and is also discussed in further detail in Section 2.4.5 of this safety evaluation.

Since welds H2 and H3 affect the shroud stiffness, a special case of crack separation during normal/upset operation and accident conditions was investigated in a supplemental analysis (licensee's submittal of February 28,

1994) whereby throughwall 360° cracking was postulated simultaneously at H2 and H3. The analysis does not postulate cracking at H8, but covers cracking at all other welds (H1 - H7). The results of the H8 weld inspections validate the assumption that the H8 weld is highly unlikely to experience a 360° throughwall crack (See Section 2.5.1). An ANSYS finite element model was prepared that included details at the top guide support ring and at the conical support. The stabilizer stiffness and the stiffness of the lower support are also included in the preload calculations and the supplemental stress evaluation. Welds H2 and H3 are full-penetration welds with a 0.63 fillet on the ring side. The following four cases were evaluated by the licensee since they were considered to be bounding in determining the stiffness at the top guide ring as a result of various postulated cracking scenarios.

Case 1. Welds H2 and H3 have a 360° throughwall crack on the ring side of the fillet weld.

Case 2. Welds H2 and H3 have a 360° throughwall crack on the shroud shell side of the fillet weld.

Case 3. Welds H2 and H3 have a 360° throughwall crack with no fillet weld remaining.

Case 4. Welds H2 and H3 are not cracked.

Metallurgical evidence from reactor weld failures analysis suggest Case 1 is the most likely to occur for cracks extending greater than 180°. Cases 1 through 3 bound the ring stiffness for the postulated crack scenarios.

During normal operation at 105% core flow, the core support pressure drop, is 15.9 psi and the shroud head pressure drop is 5.9 psi. The calculated lift load was found to be less than the estimated compressive load at welds H6B and H7. The results for all other cases considered also indicate that the compressive thermal preload plus weight of the internals exceeds the magnitude of the load required to separate the welds. On this basis, crack separation is not projected to occur during normal operation.

During a main steam line break accident condition, the loads on the stabilizers can exceed the thermal preload and there may be a brief separation at postulated crack locations. The most severe conditions are 360° throughwall cracks at welds H6B, H7 or H8. Failure at one or more of these welds transfers the loads due to pressure differential across the core to the stabilizers which, when combined with a seismic event loads, will result in a brief maximum separation at the weld H6B of about 0.63 inches. This displacement is temporary since the stabilizers will springback and the weight of the internals is sufficient to close the gap once the event is over. Lateral motion is restricted by the stabilizer springs and clamps/spacers. In the course of review of the analysis relating to the crack separation during normal/upset operation and faulted conditions, the staff requested additional calculational details to support the load development, analytical results and assumptions. Based on its review as discussed above, the staff finds that the (proprietary) methodology to evaluate crack separation under normal operation and postulated accident conditions is acceptable and the resulting cracks do not violate the generic VIP shroud repair guidelines. The impact of leakage from the estimated cracks is discussed in Sections 2.4.4 and 2.4.5 of this safety evaluation.

2.3.3 Seismic Analysis

The seismic analysis performed by the licensee is addressed in the document entitled "Seismic Design Report of Shroud Repair for Nine Mile Point 1 Nuclear Power Plant" GE-NE-B13-01739-04 Rev. 0. The mathematical model used for the analysis included the reactor building, shield wall/pedestal, RPV, reactor internals, and the repair modification hardware. The structural modelling data were obtained from the information contained in the UFSAR, licensing basis calculations/reports, and design drawings. The model was analyzed using the SAP4G07 computer program discussed in the GE document NEDO-10909, Rev. 7, "SAPG07, Static and Dynamic Analysis of Mechanical and Piping Component by Finite Element Method."

An axisymmetric, lumped mass model of the RPV and internals was constructed incorporating the masses and structural properties of the various structural components. Hydrodynamic masses were calculated and included in the model to account for the dynamic coupling of the fluid mass with the solid mass. The stiffness properties of the repair modification hardware (top/bottom springs and tie rods) were incorporated in the model. The model is axisymmetric and included the equivalent rotational stiffness offered by the tie rod system.

The top and bottom lateral spring stiffnesses were incorporated in the model at the top guide and bottom core plate locations respectively.

The licensing basis horizontal Design Basis Earthquake load (DBE) is documented in the NMP-1 Design Criteria Document (DCD-115). A synthetic time history with a zero period acceleration (ZPA) of 0.11g was generated based on the horizontal DBE spectra. This time history load was used as the DBE load in this seismic analysis. Vertical seismic inertia load was not evaluated in the computer analysis in accordnace with the design basis for this facility. Vertical ZPA was calculated from the horizontal ZPA (2/3 x 0.11 = 0.073g), and was included in the analysis as a multiplier of the deadweight effects.

Consistent with the licensing basis, DBE was the only seismic load evaluated. The DBE results were used for upset, as well as emergency and faulted conditions. Ground acceleration transient response analysis by modal superposition method was used for the time history analysis.

Analysis iterations were performed to reflect the scenarios wherein 360° through-wall, circumferential cracks were postulated at the various weld locations in the shroud, including uncracked and all-welds-cracked conditions. The cracks were represented as hinges or rollers depending upon the assumed crack condition and the loading event. For an upset condition wherein the crack does not separate, the crack plane was modeled as a hinge (i.e., with no moment resistance at the crack plane). For an emergency or faulted event involving LOCA, the possibility of the shroud lifting momentarily at the crack plane exists. Under such conditions, the crack plane was modeled as a roller (i.e., with no lateral shear or moment resistance at the crack plane). Nine such governing cracked scenarios were evaluated including the uncracked case, resulting in maximum loads and displacements for the repair modification hardware design.

The maximum permanent horizontal deflection of the shroud that is not directly supported by either the upper or lower springs is limited to 0.75 inches by mechanical limit stops. In the unlikely scenario that welds H4 and H5 become loose and a combined LOCA plus seismic event occurs, the stops serve to limit the horizontal displacement to 0.75 inches, which is equal to one-half of the shroud wall thickness. These stops do not significantly affect the validity of the linear seismic analysis.

The licensing basis condition was simulated by additionally analyzing the model without the tie rod/spring modifications and without any cracks, to form a benchmark run. The resultant component loads based on the current shroud repair seismic analysis were compared with those of the benchmark run. The comparison showed insignificant changes in the results. The loads in the internal components reduce once the cracks occur. This is due to the fact that as the shroud rigidity is decreased, the fuel is isolated, and the seismic load is mainly carried by the stabilizer springs and the tie rods.

Based on its review as discussed above, the staff finds the seismic analysis methods in accordance with NRC's Standard Review Plan (NUREG-0800) and is therefore acceptable.

2.3.4 Impact of Mislocated Tie-Rods

The NMP1 Core Shroud Repair was designed with four tie rods to be located/oriented at 90, 170, 270, and 350° on the shroud support cone. However, during installation, the tie rod hole at the 170° location was made at the 166°location (i.e., 7 1/2 inches toward the 90° location). Niagara Mohawk performed an analysis of the effects of the mislocated tie rod and concluded that the shroud repair is acceptable as installed. This evaluation was provided as an attachment to their letter of March 14, 1995. Analyses performed to determine the impact on the previous seismic loads, the tie rod pressure load distribution, and the vertical displacements have been reviewed by the staff. The original governing maximum seismic loads for the tie rods, top and bottom springs were not exceeded. The maximum tie rod pressure load is increased by 3.6% with the revised stresses remaining below allowables for normal, upset, emergency and faulted conditions. The mislocation had no impact on the conclusion that no weld separation occurs for the normal condition. The maximum upset condition separation for Case 2 (See Sec. 2.3.2 of this SE) is unchanged and the Case 3 maximum separation is increased by a maximum of .002 inches to .032 inches. The maximum accident separation increases by 0.02 inches from .63 inches to .65 inches. The staff has reviewed the impact of the mislocation of the stabilizer assembly on the original shroud repair design reported in GE-NE-BB-01739-05, Rev. 1 of January 1995. The staff conclusions based on a review of this and related documents remain unchanged.

2.3.5 Potential for Flow-Induced Vibration Damage

The repair has been designed to address the potential for flow-induced vibration (FIV) and that it remains at an acceptable level. The natural frequency of the repaired shroud, including the repair hardware, has been determined. The vibratory stresses were shown to be less than the allowable stresses of the repair materials. Forcing functions considered included the coolant flow and the vibratory forces transmitted via the end point attachments for the repair. Testing used as an alternative, or to supplement the vibration analysis is addressed in the proprietary version of GE-NE-B13-01739-05, Rev. 1. The vortex shedding frequency has been shown to be well below 27Hz which is the lowest natural frequency of the stabilizer assembly. This combination satisfies the standard GE design goal of a factor of three between excitation frequency and lowest natural frequency. Therefore, the staff has concluded that FIV has no impact on the repair hardware or other reactor internals, such as the incore instrumentation.

The transients described in the NMP1 FSAR Chapter XV were reviewed. The bounding upset thermal event for the tie rod assembly is considered to be an upset condition wherein cold water is introduced into the annulus while the reactor inlet plenum remains at 543°F. This situation could potentially occur with the loss of feedwater followed by restoring the feedwater flow, but without heating. The thermal effects of this event on the shroud and the hardware have been reviewed by the staff and found to be acceptable.

2.3.6 Evaluation of Existing Internal Components Impacted by Repair

Stresses on the original structure of the shroud, which are directly impacted by the shroud repair hardware, have been demonstrated to be acceptable. The results of this evaluation are documented in GE Report NE-24A6426, Rev. 1, "Reactor Pressure Vessel Stress Report" and the licensee submittal GE-WE-B13-01739-04, Rev. B, "Shroud Repair Hardware Stress Analysis" for all of the postulated accidents.

For normal operating conditions, the preload on the tie rods will be carried by the shroud at four locations approximately equally spaced around the circumference. The stress levels on the welds H1 through H8 are bounded by the conditions occurring at weld H8. The results of the analysis on weld H8 demonstrate that the maximum impact of the installed tie rod during normal operating conditions on stress intensity is approximately 0.04% (increase in total stress intensity) or -6.44% (decrease in membrane + bending stress intensity). The membrane stress intensity decreases by 6.22%. With the exception of the total stress intensity that increases very slightly on one surface, all stress intensities drop a small amount as a result of tie rod preload. This impact is considered to be minimal and therefore verifies that the tie rod has an insignificant impact on the existing welds (H1 through H8). Stresses on the supporting structure of the shroud, which are directly impacted by the shroud repair hardware, have been demonstrated to be acceptable. The staff therefore finds the effect of the repair hardware on existing components acceptable from a structural standpoint. The stresses on the supporting structure of the shroud which are directly impacted by the shroud repair hardware have also been demonstrated to be acceptable.

2.3.7 Loose Parts Considerations

Repair hardware mechanical components have been designed to minimize the potential for loose parts inside the vessel. The design repair uses mechanical locking methods (such as crimped jam nuts) for threaded connections. All parts are captured and held in crimping that is designed to last for the design life of the repair. The repair hardware is fabricated from stress corrosion resistant material. Therefore, the likelihood of a component failure is fairly remote. However, if one stabilizer is postulated to fail during normal plant operation, there would be no consequence to the shroud (even if it is cracked) or to the other three stabilizers. Potential for damage from loose parts generated by the repair and tooling operations, such as the very fine debris resulting from Electrical Discharge Machining (EDM) also referred to as "swarf," has been evaluated. The staff has reviewed the discussion provided in the proprietary version of the licensee's submittal (GE-NE-B13-01739-05, Rev. 1). On the basis of its review, the staff concludes that the EDM, metal, and honing particles generated by the installation operations do not represent a concern for fuel fretting, seal wear or instrumentation damage.

2.3.8 Evaluation of the Deviations During Installation

In the course of a post-installation inspection of the shroud repair, the licensee identified three deviations that were subsequently evaluated. These are documented in the licensee's submittals, NMP1L-0927, of March 23, 1995.

The first deviation relates to the clearance between the tie rod mid-support and the shroud. According to the design specification it should have been 0.75 inches. However, the gap between the shroud and mid-support was found to be less than 0.75 inches. Based on a review of the analysis relating to this condition, the staff finds that the original seismic analysis remains valid; however, a contact between the shroud and mid-support during faulted event could potentially occur. The stresses in the shroud, hardware and reactor pressure vessel resulting from this possible contact were found to be acceptable. The staff, therefore, concludes that there is no adverse impact due to this deviation.

The second deviation pertains to the positioning of the lower stabilizer spring contact. The spring contact should have been located between the H5 and H6A welds. However, the inspections revealed that the spring contact was actually located slightly below the H6A weld at all four tie rod locations. As a result, the barrel section between the H5 and H6A welds would not be laterally restrained during a main steam line LOCA combined with a DBE as was originally intended. The normal, upset, emergency and faulted events were reviewed by the staff to evaluate the effects of this condition. The evaluation indicated that all design-basis load combinations are met. The main steam line LOCA combined with a DBE, which is outside the NMP1 licensing basis, required additional evaluation. The evaluation of the main steam line LOCA plus DBE confirmed that the horizontal displacement of the core plate during this event will remain less than the allowable permanent core plate displacement. On this basis, the staff finds that the continued operation through the next cycle is justified. The licensee will implement appropriate corrective actions by the end of the next refueling outage. The staff will review the proposed corrective actions prior to implementation.

The third deviation concerns the lower spring wedge which bears against a recirculation nozzle weld at the 270° location. The inspection indicated that the contact area between the lower wedge and the reactor pressure vessel wall is approximately 2/3 of the wedge area. This condition was evaluated considering the potential for wedge rotation or sliding at the contact surface due to hydraulic asymetric loads and the load on the nozzle. As a result of its review, the staff finds that all existing analyses remain valid. The flow velocity in this region is less than the velocity directly in front of the nozzle which was used in the original flow-induced vibration analysis. Therefore, the existing flow-induced vibration analysis remains valid.

2.4 SYSTEMS EVALUATION

2.4.1 Tie-Rod System-Induced Leakage

The installation of the tie-rod assemblies requires the machining of eight holes in the shroud head flange and eight holes in the shroud support cone. The licensee also planned for the installation of the H8 weld brackets which would require the machining of twenty four holes in the lower shroud. The licensee estimates that a small amount of core flow leakage through the clearance between the holes and the mating bolts and shear keys will occur. The total calculated leakage from the installation of the tie-rod assemblies and H8 brackets was estimated to be 0.70% of core flow at 100% rated power and 85 to 100% rated core flow. Although this leakage is not significant with regards to total core flow and would be acceptable by the staff, the staff noted that the leakage rate would be reduced with only the installation of either the tie-rod assemblies or the H8 brackets. By letter dated February 28, 1995, NMPC informed the staff that the installation of brackets at the H8 weld is not necessary based on the results of the ultrasonic examination of the H8 weld. Therefore, with only the tie-rod assemblies installed, the total calculated leakage was estimated to be 0.33% of core flow at 100% rated power and 85 to 100% rated core flow. The staff does not consider this leakage rate to be significant with regards to total core flow and therefore, is acceptable.

At NMP1, the ECCS consists of the single-train feedwater coolant injection (FWCI) system, the automatic depressurization system (ADS), and the two-train core spray (CS) system. The FWCI system requires limited offsite power to be functional. During a LOCA, the core spray system transfers water from the suppression pool to the reactor vessel where the water cools the core and returns to the suppression chamber via the break. Based on the above description of the core spray, the staff notes that leakage through the clearance of the repair holes does not affect the performance of the core spray system. Therefore, ECCS performance is not affected by the physical installation of the tie-rod system and/or the H8 weld brackets.

2.4.2 Shroud Weld Crack Leakage

The tie-rod assemblies are installed with a cold preload to ensure that no vertical separation of any or all cracked horizontal welds will occur during normal operations. Vertical separation, if sufficiently large, could compromise fuel geometry and control rod insertion. For NMP1, a maximum vertical separation of 13.3 inches is required for the top guide to clear the top of the fuel channels. With the repair, the licensee stated that the preload on the tie-rods will not allow vertical separation of failed welds during normal operations. The staff notes that, with or without the repair, the estimated vertical separation during normal operations will not affect the fuel geometry, and therefore, control rod insertion is not precluded. However, a small leakage path could exist due to existing through-wall shroud weld cracks. The licensee conservatively modeled the crack to provide a 0.001 inch leakage path per weld. The leakage through the postulated shroud cracks was determined to be approximately 10 gpm for cracks above the core plate, and 20 gpm for cracks below the core plate. The total leakage from all welds, H1 through H8, having 360° through-wall cracks was approximately 120 gpm. Although shroud crack leakage is unlikely due to the preload on the tie-rod, the licensee concluded that there are no consequences associated with the repair installed based on these small leakages during normal operations. The staff acknowledges that the total leakage is insignificant and will not affect the performance of the ECCS.

2.4.3 Downcomer Flow Characteristics

The licensee analyzed the available flow area in the downcomer with the four tie-rod assemblies installed. The staff reviewed downcomer flow calculations for the upper and lower annulus area which accounted for the core spray piping, the upper support and spring, and the lower spring and C-spring. The licensee's calculations demonstrated that the installation of the tie-rod assemblies will decrease the available downcomer flow area by 5.3 percent in the upper annulus region and 3.3 percent in the lower annulus region. Due to the small diameter of the tie-rods, the decrease in available flow area in the middle region of the annulus was approximately 0.4 percent. Based on the licensee's analysis, the staff concluded that the installation of the tie-rod assemblies will not have a significant impact on the downcomer flow characteristics. Although the licensee did not provide the corresponding pressure drop to the decrease in downcomer flow area, the staff concluded that the pressure drop is insignificant based on other reviews of similar core shroud repairs. Therefore, the staff agrees with the licensee that the installation of the tie-rod assemblies should not affect the recirculation flow of the reactor.

2.4.4 Potential Lateral Displacement of the Shroud

The licensee also evaluated the maximum lateral displacement of the shroud at the core support plate and upper guide plate under normal operations and load combinations such as design basis earthquake (DBE), main steam line break (MSLB), and recirculation line break (RLB). Lateral displacement of the shroud could damage core spray lines and could produce an opening in the shroud, inducing shroud bypass leakage and complicating recovery. Lateral seismic restraints have been included in the proposed design which will limit the lateral displacement of the shroud to 0.75 inches for normal and worst case accident scenarios. This lateral displacement is less than the 1.5 inch thickness of the shroud, and accordingly, the separated portions of the shroud would remain overlapped during worst case conditions. Therefore, the staff has concluded that the maximum lateral displacement of the core shroud would not result in significant leakage from the core to the downcomer region following an accident scenario.

The staff also reviewed the licensee's RLB blowdown load calculations and their affect on the potential for lateral displacement of the shroud. The licensee calculated the RLB break flow with the TRACG code based on low temperature fluid conditions. The calculated break flow was then applied to a two-dimensional potential flow theory model. Previously, the staff has not accepted loads calculated by the potential flow theory based on the lack of information to benchmark the theory and the utilization of a non-conservative assumption about the jet pumps. Since NMP1 is a non-jet pump plant, the staff's second concern does not apply. NMP1's sister plant, Oyster Creek, calculated its RLB blowdown loads using the Computational Fluid Dynamics (CFD) code COMPACT 3-D, which is capable of solving the Navier-Stokes equations in three dimensions. Comparison of Oyster Creek's and NMP1's calculated blowdown loads and input parameters established that NMP1's results are consistent with Oyster Creek's calculations. Additionally, a scoping calculation using the potential flow model was performed by the staff that included flow area blockages and head losses due to the tie-rod assemblies. This calculation provided loads comparable to Oyster Creek and NMP1.

By letter dated March 14, 1995, NMPC provided the staff with General Electric (GE) Nuclear Energy's TRACG asymmetric load calculation for NMP1. The TRACG calculation was performed with and without the tie-rods installed in order to provide validation of the potential flow methodology used. The TRACG results are more exact representations of the flow, pressures, and forces due to the RLB. The licensee compared the TRACG results without the tie-rods installed to their original potential flow model results. The comparison demonstrated that the potential flow calculation provided higher loads for nearly all elevations. This result was obtained by using the maximum break flow observed in TRACG model as the steady state break flow in the potential flow model. Further analysis of the referenced TRACG model revealed that several improvements to the potential flow model, such as increased break flow with lower feedwater temperature, increased recirculation suction nozzle internal diameter to correspond with plant as-built information, narrowed annulus area near the shroud head, and adjustment of the static pressure near the suction nozzle, could be made. The licensee made the above changes to their potential flow model and calculated the additional force due to the four tie-rods. The staff has reviewed the new potential flow model blowdown loads and concluded that they are conservative. Potential lateral displacement of the shroud following an RBL with the new blowdown loads is still limited to 0.75 inches by the mechanical stops. Therefore, the staff concluded that NMP1's RLB blowdown loads are acceptable.

As stated earlier, on March 7, 1995, the licensee informed the staff that one tie-rod assembly was installed at the wrong location, i.e. 166° instead of 170°. The staff evaluated the affect of the different location with regards to bypass leakage and potential horizontal shroud displacement. Since the same size bolt holes were machined into the shroud head flange and support cone at the incorrect location, the total bypass leakage should remain the same. Furthermore, the 4° differential does not significantly affect the potential lateral loads and horizontal shroud displacement. Therefore, the staff concluded that the installation error of the one tie-rod assembly will not affect the systems aspects of the repair.

2.4.5 Potential Vertical Separation of the Shroud

The licensee evaluated the maximum vertical separation of the shroud assuming 360° through-wall cracks at H1 through H6B during a MSLB and a MSLB plus a seismic event. These postulated events would result in a large upward load on the shroud which could impact the ability of the control rods to insert and the ability of the core spray system to perform its safety function. As stated above, a maximum vertical separation of 13.3 inches is required for the top guide to clear the top of the fuel channels. In the September 26, 1994 letter, the licensee calculated that the maximum vertical separation would be 12.1 inches during a MSLB, assuming 360° through-wall weld failure of the H3 weld location without the repair installed. With the tie-rod assemblies installed and the mislocation of one tie-rod by 4°, the maximum vertical separation is limited to 0.65 inches during the MSLB plus seismic event and significantly lower for a MSLB. This separation is limited by the tie-rods and should not impact the core spray system. The staff acknowledges that the ECCS performance and control rod insertion should not be impacted by this momentary vertical separation. Therefore, based on this assessment, the staff concluded that postulated separation during a MSLB combined with a seismic event would not preclude any systems from performing their safety functions.

The staff has evaluated the licensee's safety evaluation of the consequences of the proposed core shroud repair. The staff has found that the proposed repair should not impact the ability to insert control rods, and the performance of the ECCS, particularly the core spray system. The staff concluded that the proposed repair does not pose adverse consequences to plant safety, and therefore, plant operation is acceptable with the proposed core shroud repair installed.

2.5 MATERIALS AND FABRICATION CONSIDERATIONS

The licensee has selected Type 316 or 316L austenitic stainless steel and nickel-based (NI-CR-Fe) alloy X-750 materials for the fabrication of shroud stabilizer components. These materials have been used for a number of other components in the BWR environment and have demonstrated good resistance to stress corrosion cracking by laboratory testing and long term service experience. Welding is not designed in the fabrication and the installation of the shroud stabilizers for the purpose of minimizing its susceptibility to intergranular stress corrosion cracking (IGSCC). The components of upper and lower springs, upper nuts, upper and lower brackets, lower bracket nuts and toggle bolts will be made from alloy X-750; and the tie rods, core plate wedges and other remaining components in the stabilizer assemblies will be made from either Type 316 or Type 316L austenitic stainless steel. The licensee stated that the selected materials and fabrication methods for NMP1 shroud stabilizers are consistent with that used for the Hatch Unit 1 core shroud repair.

Both alloy X-750 and Type 316 or 316L austenitic stainless steel are acceptable ASME Code Section III materials. The alloy X-750 will be procured to American Society for Testing and Materials (ASTM) Standard B637, Grade UNS N07750 material (bars and forging) requirements with a maximum cobalt content not to exceed 0.090%. The heat treatment of alloy X-750 shall include solution annealing at 1975 ±25°F for 60 to 70 minutes and age hardened at 1300 ± 15°F for a minimum of 20 hours. Air cooling is the specified cooling method after annealing or age hardening. Equalization heat treatment at 1500°F to 1800°F is prohibited because this heat treatment will produce a microstructure that would make the material susceptible to IGSCC.

The Type 316 or 316L austenitic stainless steel will be procured to ASTM A-479, A-182 or A240 with a maximum carbon content of 0.020%. The maximum hardness of this material is limited to Rockwell B 92 for types 316 or 316L. All procured Types 316 or 316L materials are required to be tested for sensitization in accordance with ASTM Standard A262, Procedures A or E to ensure the materials were not sensitized. The components made of this material will be in a solution annealed condition. Water quenching is specified for cooling from solution annealing at 2000°F ±100°F. Certain parts are specified on the drawings to be re-solution annealed after final machining such as the machined threads of the tie rods. The tie rod threads are required to be induction annealed after final machining to remove the surface cold work effect. The cold work resulting from machining is known to promote IGSCC. The licensee stated that re-solution annealing will not be applied to alloy X-750 machined surfaces because GE's metallurgical investigations have shown that their surfaces will not be affected by machining.

In the fabrication specification 25A5584, Revision 2, Section 3.2.2.1 (Austenitic 300 SST Heat Treatment) and in the SE of GE core shroud repair design (GE-NE-B13-01739-05, Revision 1), Part A.2 Materials, GE stated that the successful completion of the sensitization testing (ASTM A262, Practice A or E) shall be accepted as evidence of the correct solution heat treatment and water quenching if time and temperature charts and water quenching records are not available.

To ensure there is no intergranular attack as a result of high temperature annealing or pickling treatment, the licensee requires IGA testing per GE E50YP11 specification to be performed for each heat and heat treat lot of materials after annealing or pickling. IGA testing is not required if a minimum of 0.030 inches of material is removed from all surfaces of the product after final annealing or pickling.

The licensee indicated that stabilizer parts are generally rough machined to within 0.10 inch of final size and skim passes are used to achieve the final dimensions. Coolant and sharp tools will be used in machining. The final machined surface finish is specified to be 123 root mean square or better.

The licensee also indicated that the thread lubricant D50YP5B will be used in the installation of stabilizer assemblies. Controls of lubricant impurities are provided in the GE Specification D50YP12, where impurities limits are specified for halogens, sulfur, nitrates and low melting point metals.

The staff has reviewed the licensee's submittal regarding the proposed core shroud repair and concludes that the selected materials and fabrication methods for the stabilizer assemblies are acceptable.

2.5.1 Pre-Modification and Post-Modification Inspection

The licensee's pre-modification inspection plan to support the repair installation consists of inspection of circumferential welds H-8 and H-9 and certain vertical welds and top ring segment welds. The selection of the welds and the scope of the inspection are briefly summarized below:

(1) Enhanced visual examination of the H-9 weld at four locations adjacent to the tie rods with a minimum of 26 inches in length at each location. The 26 inches weld length includes the weld length adjacent to the two toggles (12 inches) and an additional 7 inches of weld length at each end for stress attenuation. The weld H-9 connects the core support cone to the reactor pressure vessel and is a part of the load path from the tie rods to the reactor pressure vessel;

(2) Volumetric examination of H-8 weld of all accessible areas and supplemented with enhanced visual examination. The H-8 weld is a

dissimilar metal weld which connects the core support cone (alloy 600) to the core shroud (sensitized Type 304 stainless steel forging). The H-8 weld provides vertical support to the core shroud;

(3) Enhanced visual examination from inside surface of four (4) core shroud vertical welds (V9, V10, V11 and V12). These vertical welds intersect the H-5 circumferential weld and each weld will be examined a section of six (6) inches in length. The H-5 weld is located in the vessel beltline region which is subject to higher radiation exposure than at any other weld location. The hoop stresses in the shroud cylinder are low and the required sound vertical weld to support the design repair is very minimal;

(4) Enhanced visual examination of the accessible areas of the top guide ring segment welds V5 and V6 from the top inside surface. The structural integrity of the top guide support ring is essential to the maintaining of the required preload in the tie rods.

The licensee stated that the inspection was performed and its techniques qualified in accordance with the guidelines delineated in BWRVIP documents "BWRVIP Standards for Visual Inspection of Core Shrouds" and "BWRVIP Core Shroud NDE Uncertainty and Procedure Standards." Ultrasonic examination (UT) was performed on H-8 weld using a 45 degree shear, 60 degree refractive longitudinal and OD creeping wave transducers. The licensee reported that the UT examination successfully inspected about 45% (260 inches) of the weld circumference from four quadrants of the H-8 weld. Due to the access limitation at weld H8, 18% of the total volume would not be covered by the UT examination. Additional 27% of the weld circumference (160 inches) was visually examined above the H-8 weld on the vertical surface of the shroud support ring using a camera capable of resolving a 0.005 inch wire against a neutral gray background.

A single UT indication was found on the underside of the shroud support cone. This indication was located at the interface of the lower weld (Inconel 182) and the base material (alloy 600). The size of the indication was reported to be 0.5 inches in depth (about 33% through wall) and about 3.12 inches in length. The licensee performed the root cause evaluation and concluded that the subject crack was likely to be initiated from a lack of fusion weld site (alloy 182) and propagated into the alloy 600 conical support base material. The cracking mechanism is presumed to be intergranular stress corrosion cracking (IGSCC). Since the length of the crack indication is short (less than 1.5% of the total inspected length), the licensee concludes that the subject crack indication is not structurally significant.

Five (5) small indications with length varying from 0.5 to 0.75 inch were found by enhanced visual examination on the vertical surface of the shroud support ring. The shroud support ring was made of stainless steel Type 304 forging and was furnace sensitized during heat treatment of the vessel. These indications are very tight exhibiting IGSCC characteristics. Four of the five indications are grouped within an area between the azimuths 348 degrees through 356 degrees. By adding the measurement uncertainties of 1.25 inches to each end, the total length of a cluster of the four indications is about

15.3 inches. The licensee concludes that the crack growth of this group of indications using the NRC approved bounding crack growth rate ($5x10^{-5}$ inch/hour) will result in no significant reduction in the structural margin through several cycles. The licensee stated that they will reinspect all the reported indications at the next refueling outage to confirm the postulated crack growth of these indications.

Enhanced visual examination was performed on the top surface of the H-9 weld at four (4) locations where tie rods will be installed. At each location, a circumferential length of about 26 inches was inspected. A section of 6 inches was inspected at each of the vertical welds of V9, V10 and V11 which intersected the circumferential weld H5. No crack indications were found at these weld locations. The inspection personnel could not locate the vertical weld V12 and the segment welds V5 and V6 of the top guide support ring and, therefore, inspection was not performed on these welds. The licensee stated that it is difficult to locate segment welds V5 and V6 because the support ring was machined after fabrication and welding. In searching for V5 and V6 welds, the top ring surface was cleaned and inspected for more than 180 degrees; and no degraded condition was found.

The licensee performed enhanced visual examination at four locations of H-2 and H-3 welds with each location adjacent to a repair tie rod. The area examined at each location is approximately 36 inches in length and includes both the upper and lower heat affected zones of the weld. The H-2 weld was examined from the outside diameter surface of the shroud as the examination of H-3 weld was performed from the inside diameter surface. The licensee reported that rejectable indications were found in the upper heat affected zone of H-3 weld at three of the four inspected locations. These indications were reported to exhibit intergranular stress corrosion cracking (IGSCC). The cracking essentially extends through the entire length (36 inches) of the three examined locations. The upper heat affected zone of H-3 weld is located at the inner vertical surface of the top guide support ring. The top guide support ring was made of two welded segments of rolled plates (type 304 stainless steel). The observed cracking in the support ring is consistent with the industry experience in core shroud examination. Since the integrity of H-2 and H-3 welds is not required to support the proposed core shroud repair, the future reinspection of these welds is not required.

In a response to the staff's request for additional information (RAI), the licensee stated that they will submit plans for reinspection of core shroud repair assemblies and core shroud when the BWRVIP guidelines are established. The licensee also stated that the reinspection plan of the repair assemblies will also consider the potential degradation in threaded areas and locations of crevices and stress concentration. The staff recommends that the licensee proposed reinspection plan should also consider the plant specific repair design requirements and the extent and the results of the baseline inspection performed during pre-modification inspection. The staff will review the licensee's reinspection plans for the core shroud and repair assemblies when submitted. However, the licensee should submit their reinspection plans within six months after restart from the current refueling outage. Since the core shroud and its repair assemblies are classified as ASME Code Class B-N-2 components (core structural support), the reinspection plans when approved by NRC should be incorporated into the ASME Section XI in-service (ISI) program.

The staff also recommends that the licensee should incorporate the following when performing reinspection during the next refueling outage (1) the qualification of the UT techniques should include a mock-up block which simulates the configuration of the H8 dissimilar metal weld, and (2) the development of an effective method to locate the segment welds of the top guide support ring.

The staff has reviewed the licensee's inspection results. The staff concludes that the licensee's inspection is acceptable to support the planned core shroud repair. Although some cracks were found, they are minor and would not impact the structural integrity of the welds during the operation in the next fuel cycle.

3.0 CONCLUSION

Based on a review of the shroud modification hardware from structural, systems, materials and fabrication considerations, as discussed above, the staff finds that the proposed modifications of the NMP1 core shroud are acceptable.

Principal Contributors: J. Rajan, K. Kavanagh, W. Koo

Dated: March 1995

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

March 31, 1995

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (NMP1), EVALUATION OF CORE SHROUD STABILIZER DESIGN (TAC NO. M91273)

Dear Mr. Sylvia:

The purpose of this letter is to transmit our safety evaluation (SE) of the subject matter. Based on our review, we find that your proposed core shroud stabilizer design is acceptable as documented in the enclosed safety evaluation.

By letter dated January 6, 1995, as supplemented by letters dated January 23, and 26, February 14, 24, and 28, March 7 and 9, two on March 13, two on March 14, March 23, 27, 28, and 30, 1995, Niagara Mohawk Power Corporation (NMPC) submitted the details of the planned repair of the circumferential welds for the Nine Mile Point Unit 1 (NMP1) reactor core shroud. Information was also provided to the NRC staff during conference calls held on March 1, 3, 23, 24, and 27, 1995. The March 28, 1995, letter confirmed that the information provided during the conference calls would be formally submitted on the NMP1 docket no later than March 31, 1995.

Initially, NMPC's planned permanent repair involved installation of four tie-rod assemblies combined with core plate wedges to replace welds H1 through H7 and six brackets to replace the downward vertical load capability of the H8 weld. It was NMPC's intention to examine the H1 through H8 shroud welds in accordance with the Boiling Water Reactor Vessel and Internals Project Inspection Criteria and install the tie-rod assemblies and/or the H8 weld brackets only if cracking was found to be unacceptable for continued plant operation. Based on the results of the ultrasonic examination of the H8 weld, NMPC decided to install the four tie-rod assemblies and not the brackets.

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The NRC staff has reviewed the above submittals. Our evaluation is provided in the enclosed Safety Evaluation. The proposed core shroud repair has been designed as an alternative to the requirements of ASME Boiler and Pressure Vessel Code pursuant to Title 10, Code of Federal Regulations, Part 50.55a(a)(3)(i). This alternative is acceptable. This completes our action with respect to TAC No. M91273.

Sincerely,

LB Marsh

Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosure: Safety Evaluation

cc w/encl: See next page

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

SAFETY EVALUATION OF THE REPAIR PROPOSAL

FOR THE NINE MILE POINT UNIT NO. 1 CORE SHROUD

NIAGARA MOHAWK POWER CORPORATION

DOCKET NUMBER 50-220

1.0 BACKGROUND

In Boiling Water Reactors (BWRs) the core shroud is a stainless steel cylinder within the reactor pressure vessel (RPV) that provides lateral support to the fuel assemblies. The core shroud also serves to partition feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core.

In 1991 cracking of the core shroud was visually observed in a foreign BWR. The crack in this BWR was located in the heat affected zone of a circumferential weld in the mid shroud shell. The General Electric Company (GE) reported the cracking found in the foreign reactor in a Rapid Information Communication Services Information Letter (RICSIL) 054. GE identified the cracking mechanism as intergranular stress corrosion cracking (IGSCC).

A number of domestic BWR licensees have recently performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic experience. The cracking reported in the Brunswick Unit 1 core shroud was particularly unique since it was the first time that extensive 360° shroud cracking had been reported by a licensee in a domestic BWR. The 360° shroud crack at Brunswick Unit 1 was located at weld H3 which joins the top guide support ring to the mid shroud shell. Information Notice (IN) 93-79 was issued by the NRC on September 30, 1993, in response to the observed cracking at Brunswick Unit 1.

The cracks reported by the Commonwealth Edison Company (ComEd, the licensee for the Dresden, Lasalle, and Quad Cities Units) in the Dresden Unit 3 and Quad Cities Unit 1 core shrouds were of major importance, since they signified the first reports of 360° cracking located in lower portions of BWR core shrouds. These 360° cracks are located at shroud welds H5, which join the core support plate rings to the middle shroud shells in the Dresden and Quad Cities Units. IN 94-42 and its supplement were issued by the NRC on June 7 and July 19, 1994, respectively, to alert other licensees of the shroud cracking discovered at Dresden Unit 3 and at Quad Cities Unit 1.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03 to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in the reactors' core shrouds.

Enclosure

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GL 94-03 requested BWR licensees to take the following actions with respect to the core shrouds:

- inspect the core shrouds no later than the next scheduled refueling outage;

- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;

- develop an inspection plan which addresses inspections of all shroud welds, and which delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject;

- develop plans for evaluation and/or repair of the core shroud;

- work closely with the BWROG on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to IGSCC.

By letter dated January 6, 1995, as supplemented by letters dated January 23, and 26, February 14, 24, and 28, March 7 and 9, two on March 13, two on March 14, March 23, 27, 28, and 30, 1995, Niagara Mohawk Power Corporation (NMPC) submitted the details of the planned repair of the circumferential welds for the Nine Mile Point Unit 1 (NMP1) reactor core shroud. Information was also provided to the NRC staff during conference calls held on March 1, 3, 23, 24, and 27, 1995. Initially, NMPC's planned permanent repair involved installation of four tie-rod assemblies combined with core plate wedges to replace welds H1 through H7 and six brackets to replace the downward vertical load capability of the H8 weld. It was NMPC's intention to examine the H1 through H8 shroud welds in accordance with the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Inspection Criteria and install the tie-rod assemblies and/or the H8 weld brackets only if cracking was found to be unacceptable for continued plant operation. Based on the results of the ultrasonic examination of the H8 weld (see Section 2.5.1), NMPC decided to install the four tie-rod assemblies and not the brackets.

2.0 <u>EVALUATION</u>

2.1 Scope of the Modification Design

The licensee indicated that the design life of all repair hardware is 25 years (the remaining life of the plant, plus life extension beyond the current operating license) which accounts for 20 effective full power years. The proposed modification takes into account 3, 4, or 5 recirculation pump operation, 105% core flow, and fluctuations in feedwater temperature during normal operations including loss of feedwater heating with a scram. The proposed modification is intended to maintain the structural integrity of the shroud with postulated 360° throughwall failure of welds H1 through H7. Thus, the functions of these welds is replaced with four stabilizer assemblies. The NMP1 repair of the core shroud is considered a non-American Society Mechanical

Engineers Boiler and Vessel Code (ASME Code) repair and, therefore, is performed as an alternative to the ASME Code Section XI, pursuant to 10 CFR 50.55a(a)(3).

2.2 Shroud Stabilizer Design Description

The design of the NMP1 core shroud modification consists of four sets of stabilizer assemblies, which were installed approximately 90° apart. Each stabilizer assembly consists of an upper spring, an upper bracket and tie rod support, a tie rod, a mid-span tie rod support, a lower spring, a lower anchor assembly, and other minor parts. The tie rod provides the vertical load carrying capability from the upper bracket to the lower anchor assembly attached to the RPV core shroud support cone, and provides support for the springs. The vertical locations of the radial springs were chosen to provide the maximum support for the shroud, top guide, core plate, and, the fuel assemblies. The upper spring provides radial load carrying capability from the shroud, at the top guide elevation, to the RPV. The lower spring provides radial load carrying capability from the shroud, at the core support plate elevation, to the RPV. The upper stabilizer bracket provides an attachment feature to the top of the shroud as well as restraint of the upper shroud welds. The mid-span tie rod support is installed to provide a limit stop for the shroud cylinder between the H4 and H5. The mid-span tie rod support which is preloaded against the RPV effectively divides the tie rod into two shorter, stiffer rods to increase the natural frequency of the tie rod assembly, thereby preventing unacceptable levels of flow-induced vibration. At the top, each stabilizer assembly fits through two slots, which are machined into the non-safety-related shroud head and steam surface of the shroud top flange. The assembly then extends downward to below weld H3. The stabilizer assembly supports the upper spring and has a hole through which the tie rod passes. The tie rod is held against the upper bracket with a nut. The tie rod extends downward approximately 136 inches and is threaded into the lower spring. The lower spring has a pin at the bottom, which is attached to the clevis in the lower support. The lower support is bolted to the shroud support cone with two toggle bolts. The primary forces that the stabilizers would experience are from seismic events, loss-of-coolant accident (LOCA) differential pressure loads, and differential thermal expansion. The stabilizer assemblies and cracks in the shroud change the seismic response of the reactor internals. Thus, it was necessary to modify the seismic analysis of the reactor to include the effects of the cracks and the stabilizers. This dynamic analysis was performed in an iterative manner to determine the appropriate values of the spring constants of the upper and lower springs as well as the number of stabilizer assemblies required. The analysis results discussed below indicated that four stabilizer assemblies would be acceptable.

2.3 Structural Evaluation

2.3.1 Stabilizer Assemblies

The stabilizers were designed to the structural criteria specified in the NMP1 Updated Final Safety Analysis Report (UFSAR). The UFSAR compares the calculated shroud stresses against the allowable stress (Sm) for all operating conditions and events. Allowable stress intensities for other stress combinations and accident conditions are not addressed in the UFSAR. The purchase specification for the RPV designates the following allowable stress limits. The primary membrane stress is limited to Sm, 1.5 Sm and 2.0 Sm during normal/upset, emergency and faulted events, respectively. The primary membrane plus bending stress is limited to 1.5 Sm, 2.25 Sm and 3 Sm during normal/upset, emergency and faulted events, respectively. The shear stress is limited to 0.6 Sm, 0.9 Sm and 1.2 Sm during normal/upset, emergency and faulted events, respectively. These allowable stress intensities are consistent with the allowables used in other shroud designs reviewed by the NRC staff. The staff finds these allowable stresses acceptable. All of the loads and load combinations specified in the UFSAR, that are relevant to the core shroud, were evaluated in the design. The stabilizers are installed with a small tension preload of 3,000 lbs., to ensure that all components are tight. The stabilizer assemblies will be thermally preloaded to 79,670 lbs. during normal operating conditions. This tensile load in the tie rod results from the thermal expansion coefficient for the new stabilizer hardware being less than the thermal expansion coefficient of the shroud. The maximum permanent horizontal deflection of any part of the shroud that is not directly supported by either the upper or lower radial springs is limited to approximately 0.75 inches by mechanical limit stops. These stops do not perform this function unless a section of the shroud, for example between H4 and H5, becomes loose and a combined LOCA plus seismic event occurs. If this scenario occurs, the stops will limit the horizontal displacement to approximately 0.75 inches, which is equal to one-half of the shroud wall thickness. A displacement equal to one-half of the shroud wall thickness will not result in post event leakages that prevent core cooling, because the shroud sections still overlap each other by one-half (0.75 inches) of the shroud wall thickness. In addition, control rod insertion will not be precluded by maximum lateral displacement of the core shroud (see Section 2.4.4).

Wedges between the core support and the shroud (also called the Clamp/Spacer) are required at each stabilizer location to prevent relative motion of the core plate to the shroud. The four spacers are located in the annulus between the core support and the shroud and rest on the shroud ring. The wedges are held in place by clamping under the existing angle brackets that position the existing shield blocks. The annulus is measured at each location and the spacers are machined for a maximum clearance of 0.030 inches at the core plate elevation. In the event that welds H6A and H6B should fail, the wedges would provide a direct load path from the core plate to the shroud to help distribute the lateral loads occurring during a seismic event. The shroud cylinder at this location is restrained in the lateral direction by the lower tie rod spring.

The upper and lower springs of the stabilizers are installed with a small radial preload such that they provide radial support for the shroud. During normal operation, the shroud and stabilizer springs radially expand due to thermal growth slightly more than the RPV, which increases the radial preload and assures that the springs provide lateral support for the shroud during normal operation.

The vertical locations of the upper and lower springs were chosen to provide the maximum horizontal support for the fuel assemblies. The upper springs are at the top guide elevation and the lower springs are at the core support plate elevation. All of the horizontal support for the fuel assemblies is provided by the top guide and the core support plate.

A detailed finite element model, using the COSMOS code of the NMP1 shroud and repair assembly, was developed for stress analysis purposes to fully evaluate all of the loading conditions specified in GE Design Specification No. 25A5583, Revision A, "Shroud Repair Hardware." The model consisted of a 180° shroud segment that incorporated the shroud shell, gaps (representing cracks), vertical tie rod assemblies/repair springs, and lower brackets. Repair spring and vertical tie rod assemblies were included in the model as 3-D truss elements and lower brackets as 3-D beam elements representing the repair hardware global mechanical characteristics. A 180° segment was necessitated by the need to evaluate the nonsymmetric loads.

The shroud spring and vertical tie rod components were separately modeled in detail to evaluate their mechanical characteristics and behavior. These models are described in detail in the licensee submittal GE-NE-B13-01739-04, Revision B.

The COSMOS finite element code has been verified for use in the nuclear power industry in accordance with the requirements of 10 CFR Part 50, Appendix B, and the applicable sections of ANSI/ASME QA-1 and related supplements. In the above submittal (GE-NE-B13-01739-04), the licensee indicated the COSMOS code users' guide documents a close comparison between finite element analysis results and closed form solutions for over 1000 problems of different type elements and loading conditions. For validating the COSMOS code for NMP1 application, the verification problems for the elements used in the shroud analysis (Solids, 3-D beam, rigid bar, spring, coupling and gap) were reanalyzed by the licensee.

Based on its review of the analysis presented by the licensee in its submittal, GE-NE-B13-01739-04, "Shroud Repair Hardware Stress Analysis - Nine Mile Point Unit 1" and related documents, the NRC staff finds that the maximum stresses in the tie-rods, upper and lower springs and supports including the shroud conical support remain within the allowables for applicable normal, upset and faulted conditions. Therefore, the structural integrity of the shroud and repair hardware is maintained after the proposed repairs. However, if an upset or faulted condition occurs, the licensee has committed in a letter dated March 30, 1995, to evaluate the effect of the event on the shroud and the tie rod assemblies (including the preload) prior to returning to power operation.

2.3.2 Evaluation of Postulated Critical Weld Failures

The licensee analyzed the worst-case scenario for 360° through-wall cracking in all the circumferential welds from H1 through H7. Since cracking at welds H2 and H3 could affect the shroud stiffness, and therefore the preload, additional stress analysis was performed to evaluate this condition. The results confirm that gaps would not develop under normal operating conditions for cracks at welds H1 through H7. For upset conditions, conservative assumptions predict a maximum separation of .030 inches. The existence of gaps during conditions other than normal operation does not violate the generic VIP shroud repair guidelines. The potential crack separation for upset event conditions is temporary and is projected to close following the event since the thermal preload will be recovered. The licensee's calculations indicate that the installation preload would not be affected following an upset event and that the calculated tie rod assembly stresses would remain within elastic limits. Realistic assumptions regarding the H2 and H3 fillet weld integrity demonstrate that no separation would occur for bounding 100% rated core flow upset condition pressures. In the evaluation for faulted accident conditions, gaps are predicted at several weld locations. An assessment of the consequences from this event was provided in a previous safety evaluation dated July 21, 1994, for Dresden Unit 3 and Quad Cities Unit 1, and is also discussed in further detail in Section 2.4.5 of this SE.

Since welds H2 and H3 affect the shroud stiffness, a special case of crack separation during normal/upset operation and accident conditions was investigated in a supplemental analysis (licensee's submittal of February 28, 1994) whereby throughwall 360° cracking was postulated simultaneously at H2 and H3. The analysis does not postulate cracking at H8, but covers cracking at all other welds (H1 - H7). The results of the H8 weld inspections validate the assumption that the H8 weld is highly unlikely to experience a 360° throughwall crack (See Section 2.5.1). An ANSYS finite element model was prepared that included details at the top guide support ring and at the conical support. The stabilizer stiffness and the stiffness of the lower support are also included in the preload calculations and the supplemental stress evaluation. Welds H2 and H3 are full-penetration welds with a 0.63 fillet on the ring side. The following four cases were evaluated by the licensee since they were considered to be bounding in determining the stiffness at the top guide ring as a result of various postulated cracking scenarios.

Case 1. Welds H2 and H3 have a 360° throughwall crack on the ring side of the fillet weld.

Case 2. Welds H2 and H3 have a 360° throughwall crack on the shroud shell side of the fillet weld.

Case 3. Welds H2 and H3 have a 360° throughwall crack with no fillet weld remaining.

Case 4. Welds H2 and H3 are not cracked.

Metallurgical evidence from reactor weld failures analysis suggest Case 1 is the most likely to occur for cracks extending greater than 180°. Cases 1 through 3 bound the ring stiffness for the postulated crack scenarios.

During normal operation at 105% core flow, the core support pressure drop is 15.9 psi and the shroud head pressure drop is 5.9 psi. The calculated lift load was found to be less than the estimated compressive load at welds H6B and H7. The results for all other cases considered also indicate that the compressive thermal preload plus weight of the internals exceeds the magnitude of the load required to separate the welds. On this basis, crack separation is not projected to occur during normal operation.

During a main steam line break accident condition, the loads on the stabilizers can exceed the thermal preload and there may be a brief separation at postulated crack locations. The most severe conditions are 360° throughwall cracks at welds H6B, H7, or H8. Failure at one or more of these welds transfers the loads due to pressure differential across the core to the stabilizers which, when combined with a seismic event loads, will result in a brief maximum separation at the weld H6B of about 0.63 inches. This displacement is temporary since the stabilizers will spring back and the weight of the internals is sufficient to close the gap once the event is over. Lateral motion is restricted by the stabilizer springs and clamps/spacers. In the course of review of the analysis relating to the crack separation during normal/upset operation and faulted conditions, the NRC staff requested additional calculational details to support the load development, analytical results and assumptions (letter dated March 9, 1995). Based on its review as discussed above, the staff finds that the (proprietary) methodology to evaluate crack separation under normal operation and postulated accident conditions is acceptable and the resulting cracks do not violate the generic VIP shroud repair guidelines that have been endorsed by the NRC in the SE on Boiling Water Reactor Core Shroud Repair Design Criteria dated September 29, 1994. The impact of leakage from the estimated cracks is discussed in Sections 2.4.4 and 2.4.5 of this SE.

2.3.3 Seismic Analysis

The seismic analysis (proprietary) performed by the licensee is addressed in the document entitled "Seismic Design Report of Shroud Repair for Nine Mile Point 1 Nuclear Power Plant" GE-NE-B13-01739-04, Rev. 0. The mathematical model used for the analysis included the reactor building, shield wall/pedestal, RPV, reactor internals, and the repair modification hardware. The structural modeling data were obtained from the information contained in the UFSAR, licensing basis calculations/reports, and design drawings. The model was analyzed using the SAP4G07 computer program discussed in the GE document NEDO-10909, Rev. 7, "SAPG07, Static and Dynamic Analysis of Mechanical and Piping Component by Finite Element Method."

An axisymmetric, lumped mass model of the RPV and internals was constructed incorporating the masses and structural properties of the various structural components. Hydrodynamic masses were calculated and included in the model to account for the dynamic coupling of the fluid mass with the solid mass. The stiffness properties of the repair modification hardware (top/bottom springs and tie rods) were incorporated in the model. The model is axisymmetric and included the equivalent rotational stiffness offered by the tie rod system. The top and bottom lateral spring stiffness were incorporated in the model at the top guide and bottom core plate locations, respectively.

The licensing basis horizontal Design Basis Earthquake (DBE) load is documented in the NMP-1 Design Criteria Document (DCD-115). A synthetic time history with a zero period acceleration (ZPA) of 0.11g was generated based on the horizontal DBE spectra. This time history load was used as the DBE load in this seismic analysis. Vertical seismic inertia load was not evaluated in the computer analysis in accordance with the design basis for this facility. Vertical ZPA was calculated from the horizontal ZPA (2/3 x 0.11 = 0.073g), and was included in the analysis as a multiplier of the deadweight effects.

Consistent with the licensing basis, DBE was the only seismic load evaluated. The DBE results were used for upset, as well as emergency and faulted conditions. Ground acceleration transient response analysis by modal superposition method was used for the time history analysis.

Analysis iterations were performed to reflect the scenarios wherein 360° through-wall, circumferential cracks were postulated at the various weld locations in the shroud, including uncracked and all-welds-cracked conditions. The cracks were represented as hinges or rollers depending upon the assumed crack condition and the loading event. For an upset condition wherein the crack does not separate, the crack plane was modeled as a hinge (i.e., with no moment resistance at the crack plane). For an emergency or faulted event involving LOCA, the possibility of the shroud lifting momentarily at the crack plane exists. Under such conditions, the crack plane was modeled as a roller (i.e., with no lateral shear or moment resistance at the crack plane). Nine such governing cracked scenarios were evaluated including the uncracked case, resulting in maximum loads and displacements for the repair modification hardware design.

The maximum permanent horizontal deflection of the shroud that is not directly supported by either the upper or lower springs is limited to 0.75 inches by mechanical limit stops. In the unlikely scenario that welds H4 and H5 become loose and a combined LOCA plus seismic event occurs, the stops serve to limit the horizontal displacement to 0.75 inches, which is equal to one-half of the shroud wall thickness. These stops do not significantly affect the validity of the linear seismic analysis.

The licensing basis condition was simulated by additionally analyzing the model without the tie rod/spring modifications and without any cracks, to form a benchmark run. The resultant component loads based on the current shroud repair seismic analysis were compared with those of the benchmark run. The comparison showed insignificant changes in the results. The loads in the internal components reduce once the cracks occur. This is due to the fact that as the shroud rigidity is decreased, the fuel is isolated, and the seismic load is mainly carried by the stabilizer springs and the tie rods.

Based on its review as discussed above, the NRC staff finds the seismic analysis methods in accordance with NRC's Standard Review Plan (NUREG-0800) and is, therefore, acceptable.

2.3.4 Impact of Mislocated Tie-Rods

The NMP1 Core Shroud Repair was designed with four tie rods to be located/oriented at 90, 170, 270, and 350° on the shroud support cone. However, during installation, the tie rod hole at the 170° location was made at the 166°location (i.e., 7 1/2 inches toward the 90° location). Niagara Mohawk performed an analysis of the effects of the mislocated tie rod and concluded that the shroud repair is acceptable as installed. This evaluation was provided as an attachment to their letter of March 14, 1995. Analyses performed to determine the impact on the previous seismic loads, the tie rod pressure load distribution, and the vertical displacements have been reviewed by the NRC staff. The original governing maximum seismic loads for the tie rods, top and bottom springs, were not exceeded. The maximum tie rod pressure load is increased by 3.6% with the revised stresses remaining below allowables for normal, upset, emergency and faulted conditions. The mislocation had no impact on the conclusion that no weld separation occurs for the normal condition. The maximum upset condition separation for Case 2 (See Section 2.3.2 of this SE) is unchanged and the Case 3 maximum separation is increased by a maximum of .002 inches to .032 inches. The maximum accident separation increases by 0.02 inches from .63 inches to .65 inches. The staff has reviewed the impact of the mislocation of the stabilizer assembly on the original shroud repair design reported in GE-NE-B13-01739-05, Rev. 1, of January 1995. The staff conclusions based on a review of this and related documents remain unchanged.

2.3.5 Potential for Flow-Induced Vibration Damage

The repair has been designed to address the potential for flow-induced vibration (FIV) and that it remains at an acceptable level. The natural frequency of the repaired shroud, including the repair hardware, has been determined. The vibratory stresses were shown to be less than the allowable stresses of the repair materials. Forcing functions considered included the coolant flow and the vibratory forces transmitted via the end point attachments for the repair. Testing used as an alternative, or to supplement the vibration analysis is addressed in the proprietary version of GE-NE-B13-01739-05, Rev. 1. The vortex shedding frequency has been shown to be well below 27Hz which is the lowest natural frequency of the stabilizer assembly. This combination satisfies the standard GE design goal of a factor of three between excitation frequency and lowest natural frequency. Therefore, the NRC staff has concluded that FIV has no impact on the repair hardware or other reactor internals, such as the incore instrumentation.

The transients described in the NMP1 FSAR Chapter XV were reviewed. The bounding upset thermal event for the tie rod assembly is considered to be an upset condition wherein cold water is introduced into the annulus while the reactor inlet plenum remains at 543 °F. This situation could potentially occur with the loss of feedwater followed by restoring the feedwater flow, but without heating. The thermal effects of this event on the shroud and the hardware have been reviewed by the NRC staff and found to be acceptable.

2.3.6 Evaluation of Existing Internal Components Impacted by Repair

Stresses on the original structure of the shroud, which are directly impacted by the shroud repair hardware, have been demonstrated to be acceptable. The results of this evaluation are documented in GE Report NE-24A6426, Rev. 1, "Reactor Pressure Vessel Stress Report" and the licensee submittal GE-WE-B13-01739-04, Rev. B, "Shroud Repair Hardware Stress Analysis" for all of the postulated accidents.

For normal operating conditions, the preload on the tie rods will be carried by the shroud at four locations approximately equally spaced around the circumference. The stress levels on the welds H1 through H8 are bounded by the conditions occurring at weld H8. The results of the analysis on weld H8 demonstrate that the maximum impact of the installed tie rod during normal operating conditions on stress intensity is approximately 0.04% (increase in total stress intensity) or -6.44% (decrease in membrane + bending stress intensity). The membrane stress intensity decreases by 6.22%. With the exception of the total stress intensity that increases very slightly on one surface, all stress intensities drop a small amount as a result of tie rod preload. This impact is considered to be minimal and, therefore, verifies that the tie rod has an insignificant impact on the existing welds (H1 through H8). Stresses on the supporting structure of the shroud, which are directly impacted by the shroud repair hardware, have been demonstrated to be acceptable. The NRC staff, therefore, finds the effect of the repair hardware on existing components acceptable from a structural standpoint. The stresses on the supporting structure of the shroud which are directly impacted by the shroud repair hardware have also been demonstrated to be acceptable.

2.3.7 Loose Parts Considerations

Repair hardware mechanical components have been designed to minimize the potential for loose parts inside the vessel. The design repair uses mechanical locking methods (such as crimped jam nuts) for threaded connections. All parts are captured and held in crimping that is designed to last for the design life of the repair. The repair hardware is fabricated from stress corrosion resistant material. Therefore, the likelihood of a component failure is fairly remote. However, if one stabilizer is postulated to fail during normal plant operation, there would be no consequence to the shroud (even if it is cracked) or to the other three stabilizers. Potential for damage from loose parts generated by the repair and tooling operations, such as the very fine debris resulting from Electrical Discharge Machining (EDM) also referred to as "swarf," has been evaluated. The NRC staff has reviewed the discussion provided in the proprietary version of the licensee's submittal (GE-NE-B13-01739-05, Revision 1). On the basis of its review, the staff concludes that the EDM, metal, and honing particles generated by the installation operations do not represent a concern for fuel fretting, seal wear or instrumentation damage.

2.3.8 Evaluation of the Deviations During Installation

In the course of a post-installation inspection of the shroud repair, the licensee identified three deviations that were subsequently evaluated. These are documented in the licensee's submittals, NMP1L-0927, of March 23, 1995.

The first deviation relates to the clearance between the tie rod mid-support and the shroud. According to the design specification, it should have been 0.75 inches. However, the gap between the shroud and mid-support was found to be less than 0.75 inches. Based on a review of the analysis relating to this condition, the NRC staff finds that the original seismic analysis remains valid; however, a contact between the shroud and mid-support during faulted event could potentially occur. The stresses in the shroud, hardware and reactor pressure vessel resulting from this possible contact were found to be within the design allowables and, therefore, acceptable. The staff, therefore, concludes that there is no adverse impact due to this deviation.

The second deviation pertains to the positioning of the lower stabilizer spring contact. The spring contact should have been located between the H5 and H6A welds. However, the inspections revealed that the spring contact was actually located slightly below the H6A weld at all four tie rod locations. As a result, the barrel section between the H5 and H6A welds would not be laterally restrained during a main steam line LOCA combined with a DBE as was originally intended. The normal, upset, emergency and faulted events were reviewed by the NRC staff to evaluate the effects of this condition. The evaluation indicated that all design-basis load combinations are met. The main steam line LOCA combined with a DBE, which is outside the NMP1 licensing basis, required additional evaluation. The evaluation of the main steam line LOCA plus DBE confirmed that the horizontal displacement of the core plate during this event will remain less than the allowable permanent core plate displacement. On this basis, the staff finds that the continued operation through the next cycle is justified. The licensee will implement appropriate corrective actions by the end of the next refueling outage. The staff will review the proposed corrective actions prior to implementation.

The third deviation concerns the lower spring wedge which bears against a recirculation nozzle weld at the 270° location. The inspection indicated that the contact area between the lower wedge and the reactor pressure vessel wall is approximately 2/3 of the wedge area. This condition was evaluated considering the potential for wedge rotation or sliding at the contact surface due to hydraulic asymmetric loads and the load on the nozzle. As a result of its review, the NRC staff finds that all existing analyses remain valid. The flow velocity in this region is less than the velocity directly in front of the nozzle which was used in the original flow-induced vibration analysis. Therefore, the existing flow-induced vibration analysis remains valid.

2.4 Systems Evaluation

2.4.1 Tie-Rod System-Induced Leakage

The installation of the tie-rod assemblies required the machining of eight holes in the shroud head flange and eight holes in the shroud support cone. The licensee also planned for the installation of the H8 weld brackets which would require the machining of twenty four holes in the lower shroud. The licensee estimates that a small amount of core flow leakage through the clearance between the holes and the mating bolts and shear keys will occur. The total calculated leakage from the installation of the tie-rod assemblies and H8 brackets was estimated to be 0.70% of core flow at 100% rated power and 85 to 100% rated core flow. Although this leakage is not significant with regards to total core flow and would be acceptable to the NRC staff, the staff noted that the leakage rate would be reduced with only the installation of either the tie-rod assemblies or the H8 brackets. By letter dated February 28, 1995, NMPC informed the staff that the installation of brackets at the H8 weld is not necessary based on the results of the ultrasonic examination of the H8 weld. Therefore, with only the tie-rod assemblies installed, the total calculated leakage was estimated to be 0.33% of core flow at 100% rated power and 85 to 100% rated core flow. The staff does not consider this leakage rate to be significant with regards to total core flow and, therefore, is acceptable.

At NMP1, the emergency core cooling system (ECCS) consists of the single-train feedwater coolant injection (FWCI) system, the automatic depressurization system (ADS), and the two-train core spray (CS) system. The FWCI system requires limited offsite power to be functional. During a LOCA, the core spray system transfers water from the suppression pool to the reactor vessel where the water cools the core and returns to the suppression chamber via the break. Based on the above description of the CS, the NRC staff notes that leakage through the clearance of the repair holes does not affect the performance of the core spray system. Therefore, ECCS performance is not affected by the physical installation of the tie-rod system and/or the H8 weld brackets.

2.4.2 Shroud Weld Crack Leakage

The tie-rod assemblies are installed with a cold preload to ensure that no vertical separation of any or all cracked horizontal welds will occur during normal operations. Vertical separation, if sufficiently large, could compromise fuel geometry and control rod insertion. For NMP1, a maximum vertical separation of 13.3 inches is required for the top guide to clear the top of the fuel channels. With the repair, the licensee stated in its submittal dated January 6, 1995, that the preload on the tie-rods will not allow vertical separation of failed welds during normal operations. The NRC staff notes that, with or without the repair, the estimated vertical separation during normal operations will not affect the fuel geometry, and, therefore, control rod insertion is not precluded. However, a small leakage path could exist due to existing through-wall shroud weld cracks. The licensee modeled the crack to provide a 0.001 inch leakage path per weld. The leakage through the postulated shroud cracks was determined to be approximately 10 gpm for cracks above the core plate, and 20 gpm for cracks below the core plate. The total leakage from all welds, H1 through H8, having 360° through-wall cracks was approximately 120 gpm. Although shroud crack leakage is unlikely due to the preload on the tie-rod, the licensee concluded that there are no consequences associated with the repair installed based on these small leakages during normal operations. The staff acknowledges that the total leakage is insignificant and will not affect the performance of the ECCS.

2.4.3 Downcomer Flow Characteristics

The licensee analyzed the available flow area in the downcomer with the four tie-rod assemblies installed. The NRC staff reviewed downcomer flow calculations for the upper and lower annulus area which accounted for the CS piping, the upper support and spring, and the lower spring and C-spring. The licensee's calculations demonstrated that the installation of the tie-rod assemblies will decrease the available downcomer flow area by 5.3 percent in the upper annulus region and 3.3 percent in the lower annulus region. Due to the small diameter of the tie-rods, the decrease in available flow area in the middle region of the annulus was approximately 0.4 percent. Based on the licensee's analysis, the staff concluded that the installation of the tie-rod assemblies will not have a significant impact on the downcomer flow characteristics. Although the licensee did not provide the corresponding pressure drop to the decrease in downcomer flow area, the staff concluded that the pressure drop is insignificant based on other reviews of similar core shroud repairs. Therefore, the staff agrees with the licensee that the installation of the tie-rod assemblies would not affect the recirculation flow of the reactor.

2.4.4 Potential Lateral Displacement of the Shroud

The licensee also evaluated the maximum lateral displacement of the shroud at the core support plate and upper guide plate under normal operations and load combinations such as DBE, main steam line break (MSLB), and recirculation line break (RLB). Lateral displacement of the shroud could damage CS lines and could produce an opening in the shroud, inducing shroud bypass leakage and complicating recovery. Lateral seismic restraints have been included in the proposed design which will limit the lateral displacement of the shroud to 0.75 inches for normal and worst-case accident scenarios. This lateral displacement is less than the 1.5 inch thickness of the shroud, and accordingly, the separated portions of the shroud would remain overlapped during worst-case conditions. Additionally, a permanent lateral displacement of the top guide or core plate to 0.75 inches will not significantly increase the scram time as demonstrated in General Electric's, "Justification of Allowable Displacements of the Core Plate and Top Guide Shroud Repair, Rev. 2," dated November 16, 1994 (proprietary). Therefore, the NRC staff has concluded that the maximum lateral displacement of the core shroud would not result in significant leakage from the core to the downcomer region following an accident scenario and would not preclude control rod insertion.

The NRC staff also reviewed the licensee's RLB blowdown load calculations and their affect on the potential for lateral displacement of the shroud. The licensee calculated the RLB break flow with the TRACG code based on low temperature fluid conditions. The calculated break flow was then applied to a two-dimensional potential flow theory model. Previously, the staff has not accepted loads calculated by the potential flow theory based on the lack of information to benchmark the theory and the utilization of a nonconservative assumption about the jet pumps. Since NMP1 is a non-jet pump plant, the staff's second concern does not apply. NMP1's sister plant, Oyster Creek, calculated its RLB blowdown loads using the Computational Fluid Dynamics (CFD) code COMPACT 3-D, which is capable of solving the Navier-Stokes equations in three dimensions. Comparison of Oyster Creek's and NMP1's calculated blowdown loads and input parameters established that NMP1's results are consistent with Oyster Creek's calculations. Additionally, a scoping calculation using the potential flow model was performed by the staff that included flow area blockages and head losses due to the tie-rod assemblies. This calculation provided loads comparable to Oyster Creek and NMP1.

By letter dated March 14, 1995, NMPC provided the NRC staff with General Electric (GE) Nuclear Energy's TRACG asymmetric load calculation for NMP1. The TRACG calculation was performed with and without the tie-rods installed in order to provide validation of the potential flow methodology used. The TRACG results are more exact representations of the flow, pressures, and forces due to the RLB. The licensee compared the TRACG results without the tie-rods installed to their original potential flow model results. The comparison demonstrated that the potential flow calculation provided higher loads for nearly all elevations. This result was obtained by using the maximum break flow observed in TRACG model as the steady state break flow in the potential flow model. Further analysis of the referenced TRACG model revealed that several improvements to the potential flow model, such as increased break flow with lower feedwater temperature, increased recirculation suction nozzle internal diameter to correspond with plant as-built information, narrowed annulus area near the shroud head, and adjustment of the static pressure near the suction nozzle, could be made. The licensee made the above changes to their potential flow model and calculated the additional force due to the four tie-rods. The NRC staff has reviewed the new potential flow model blowdown loads and concluded that they are conservative. Potential lateral displacement of the shroud following an RBL with the new blowdown loads is still limited to 0.75 inches by the mechanical stops. Therefore, the staff concluded that NMP1's RLB blowdown loads are acceptable.

As stated earlier, on March 7, 1995, the licensee informed the NRC staff that one tie-rod assembly was installed at the wrong location, i.e. 166° instead of 170°. The staff evaluated the affect of the different location with regards to bypass leakage and potential horizontal shroud displacement. Since the same size bolt holes were machined into the shroud head flange and support cone at the incorrect location, the total bypass leakage should remain the same. Furthermore, the 4° differential does not significantly affect the potential lateral loads and horizontal shroud displacement. Therefore, the staff concluded that the installation error of the one tie-rod assembly will not affect the systems aspects of the repair.

2.4.5 Potential Vertical Separation of the Shroud

The licensee evaluated the maximum vertical separation of the shroud assuming 360° through-wall cracks at H1 through H6B during a MSLB and a MSLB plus a seismic event. These postulated events would result in a large upward load on the shroud which could impact the ability of the control rods to insert and the ability of the CS system to perform its safety function. As stated above, a maximum vertical separation of 13.3 inches is required for the top guide to clear the top of the fuel channels. In the September 26, 1994, letter, the licensee calculated that the maximum vertical separation would be 12.1 inches during a MSLB, assuming 360° through-wall weld failure of the H3 weld location without the repair installed. With the tie-rod assemblies installed and the mislocation of one tie-rod by 4°, the maximum vertical separation is limited to 0.65 inches during the MSLB plus seismic event and significantly lower for a MSLB. This separation is limited by the tie-rods and would not impact the CS system. The NRC staff acknowledges that the ECCS performance and control rod insertion would not be impacted by this momentary vertical separation. Therefore, based on this assessment, the staff concluded that postulated separation during a MSLB combined with a seismic event would not preclude any systems from performing their safety functions.

The NRC staff has evaluated the licensee's safety evaluation of the consequences of the proposed core shroud repair. The staff has found that the proposed repair would not impact the ability to insert control rods, and the performance of the ECCS, particularly the CS system. The staff concluded that the proposed repair does not pose adverse consequences to plant safety, and, therefore, plant operation is acceptable with the proposed core shroud repair installed.

2.5 Materials And Fabrication Considerations

The licensee has selected Type 316 or 316L austenitic stainless steel and nickel-based (NI-CR-Fe) alloy X-750 materials for the fabrication of shroud stabilizer components. These materials have been used for a number of other components in the BWR environment and have demonstrated good resistance to stress corrosion cracking by laboratory testing and long-term service experience. Welding is not designed in the fabrication and the installation of the shroud stabilizers for the purpose of minimizing its susceptibility to IGSCC. The components of upper and lower springs, upper nuts, upper and lower brackets, lower bracket nuts and toggle bolts will be made from alloy X-750; and the tie rods, core plate wedges and other remaining components in the stabilizer assemblies will be made from either Type 316 or Type 316L austenitic stainless steel. The licensee stated that the selected materials and fabrication methods for NMP1 shroud stabilizers are consistent with that used for the Hatch Unit 1 core shroud repair, which was accepted previously by the NRC staff.

Both alloy X-750 and Type 316 or 316L austenitic stainless steel are acceptable ASME Code Section III materials. The alloy X-750 will be procured to American Society for Testing and Materials (ASTM) Standard B637, Grade UNS N07750 material (bars and forging) requirements with a maximum cobalt content not to exceed 0.090%. The heat treatment of alloy X-750 shall include solution annealing at 1975 ±25 °F for 60 to 70 minutes and age hardened at 1300 ± 15 °F for a minimum of 20 hours. Air cooling is the specified cooling method after annealing or age hardening. Equalization heat treatment at 1500 °F to 1800 °F is prohibited because this heat treatment will produce a microstructure that would make the material susceptible to IGSCC.

The Type 316 or 316L austenitic stainless steel will be procured to ASTM A-479, A-182, or A240 with a maximum carbon content of 0.020%. The maximum hardness of this material is limited to Rockwell B 92 for types 316 or 316L. All procured Types 316 or 316L materials are required to be tested for sensitization in accordance with ASTM Standard A262, Procedures A or E to ensure the materials were not sensitized. The components made of this material will be in a solution annealed condition. Water quenching is specified for cooling from solution annealing at 2000 °F ±100 °F. Certain parts are specified on the drawings to be resolution annealed after final machining such as the machined threads of the tie rods. The tie rod threads are required to be induction annealed after final machining to remove the surface cold work effect. The cold work resulting from machining is known to promote IGSCC. The licensee stated that resolution annealing will not be applied to alloy X-750 machined surfaces because GE's metallurgical investigations have shown that their surfaces will not be affected by machining.

In the fabrication specification 25A5584, Revision 2, Section 3.2.2.1 (Austenitic 300 SST Heat Treatment) and in the SE of GE core shroud repair design (GE-NE-B13-01739-05, Revision 1), Part A.2 Materials, GE stated that the successful completion of the sensitization testing (ASTM A262, Practice A or E) shall be accepted as evidence of the correct solution heat treatment and water quenching if time and temperature charts and water quenching records are not available.

To ensure there is no intergranular attack as a result of high temperature annealing or pickling treatment, the licensee requires IGA testing per GE E50YP11 specification to be performed for each heat and heat treat lot of materials after annealing or pickling. IGA testing is not required if a minimum of 0.030 inches of material is removed from all surfaces of the product after final annealing or pickling.

The licensee indicated that stabilizer parts are generally rough machined to within 0.10 inch of final size and skim passes are used to achieve the final dimensions. Coolant and sharp tools will be used in machining. The final machined surface finish is specified to be 123 root mean square or better. The licensee also indicated that the thread lubricant D50YP5B will be used in the installation of stabilizer assemblies. Controls of lubricant impurities are provided in the GE Specification D50YP12, where impurities limits are specified for halogens, sulfur, nitrates and low melting point metals.

The NRC staff has reviewed the licensee's submittal regarding the proposed core shroud repair and concludes that the selected materials and fabrication methods for the stabilizer assemblies are acceptable.

2.5.1 Pre-Modification and Post-Modification Inspection

The licensee's pre-modification inspection plan to support the repair installation consisted of inspection of circumferential welds H-8 and H-9 and certain vertical welds and top ring segment welds. The selection of the welds and the scope of the inspection are briefly summarized below:

(1) Enhanced visual examination of the H-9 weld at four locations adjacent to the tie rods with a minimum of 26 inches in length at each location. The 26 inches weld length includes the weld length adjacent to the two toggles (12 inches) and an additional 7 inches of weld length at each end for stress attenuation. The weld H-9 connects the core support cone to the reactor pressure vessel and is a part of the load path from the tie rods to the reactor pressure vessel;

(2) Volumetric examination of H-8 weld of all accessible areas and supplemented with enhanced visual examination. The H-8 weld is a dissimilar metal weld which connects the core support cone (alloy 600) to the core shroud (sensitized Type 304 stainless steel forging). The H-8 weld provides vertical support to the core shroud;

(3) Enhanced visual examination from inside surface of four (4) core shroud vertical welds (V9, V10, V11 and V12). These vertical welds intersect the H-5 circumferential weld and each weld will examine a section of six (6) inches in length. The H-5 weld is located in the vessel beltline region which is subject to higher radiation exposure than at any other weld location. The hoop stresses in the shroud cylinder are low and the required sound vertical weld to support the design repair is very minimal;

(4) Enhanced visual examination of the accessible areas of the top guide ring segment welds V5 and V6 from the top inside surface. The structural integrity of the top guide support ring is essential to the maintaining of the required preload in the tie rods.

The licensee stated that the inspection was performed and its techniques qualified in accordance with the guidelines delineated in BWRVIP documents "BWRVIP Standards for Visual Inspection of Core Shrouds" and "BWRVIP Core Shroud NDE Uncertainty and Procedure Standards." Ultrasonic examination (UT) was performed on H-8 weld using a 45 degree shear, 60 degree refractive longitudinal and OD creeping wave transducers. The licensee reported that the UT examination successfully inspected about 45% (260 inches) of the weld circumference from four quadrants of the H-8 weld. Due to the access limitation at weld H8, 18% of the total volume would not be covered by the UT examination. Additional 27% of the weld circumference (160 inches) was visually examined above the H-8 weld on the vertical surface of the shroud support ring using a camera capable of resolving a 0.005 inch wire against a neutral gray background.

A single UT indication was found on the underside of the shroud support cone. This indication was located at the interface of the lower weld (Inconel 182) and the base material (alloy 600). The size of the indication was reported to be 0.5 inches in depth (about 33% through wall) and about 3.12 inches in length. The licensee performed the root cause evaluation and concluded that the subject crack was likely to be initiated from a lack of fusion weld site (alloy 182) and propagated into the alloy 600 conical support base material. The cracking mechanism is presumed to be IGSCC. Since the length of the crack indication is short (less than 1.5% of the total inspected length), the licensee concludes that the subject crack indication is not structurally significant.

Five (5) small indications with length varying from 0.5 to 0.75 inches were found by enhanced visual examination on the vertical surface of the shroud support ring. The shroud support ring was made of stainless steel Type 304 forging and was furnace sensitized during heat treatment of the vessel. These indications are very tight exhibiting IGSCC characteristics. Four of the five indications are grouped within an area between the azimuths 348 degrees through 356 degrees. By adding the measurement uncertainties of 1.25 inches to each end, the total length of a cluster of the four indications is about 15.3 inches. The licensee concludes that the crack growth of this group of indications using the NRC approved bounding crack growth rate ($5x10^{-5}$ inch/hour) will result in no significant reduction in the structural margin through several cycles. The licensee stated that they will reinspect all the reported indications at the next refueling outage to confirm the postulated crack growth of these indications.

Enhanced visual examination was performed on the top surface of the H-9 weld at four (4) locations where tie rods will be installed. At each location, a circumferential length of about 26 inches was inspected. A section of 6 inches was inspected at each of the vertical welds of V9, V10, and V11 which intersected the circumferential weld H5. No crack indications were found at these weld locations. The inspection personnel could not locate the vertical weld V12 and the segment welds V5 and V6 of the top guide support ring and, therefore, inspection was not performed on these welds. The licensee stated that it is difficult to locate segment welds V5 and V6 because the support ring was machined after fabrication and welding. In searching for V5 and V6 welds, the top ring surface was cleaned and inspected for more than 180 degrees; and no degraded condition was found.

The licensee performed enhanced visual examination at four locations of H-2 and H-3 welds with each location adjacent to a repair tie rod. The area examined at each location is approximately 36 inches in length and includes both the upper and lower heat affected zones of the weld. The H-2 weld was examined from the outside diameter surface of the shroud as the examination of H-3 weld was performed from the inside diameter surface. The licensee reported that rejectable indications were found in the upper heat affected zone of H-3 weld at three of the four inspected locations. These indications were reported to exhibit IGSCC. The cracking essentially extends through the entire length (36 inches) of the three examined locations. The upper heat affected zone of H-3 weld is located at the inner vertical surface of the top guide support ring. The top guide support ring was made of two welded segments of rolled plates (type 304 stainless steel). The observed cracking in the support ring is consistent with the industry experience in core shroud examination. Since the integrity of H-2 and H-3 welds is not required to support the proposed core shroud repair, the future reinspection of these welds is not required.

In a response to the NRC staff's request for additional information (RAI), the licensee stated that they will submit plans for reinspection of core shroud repair assemblies and core shroud when the BWRVIP guidelines are established. The licensee also stated that the reinspection plan of the repair assemblies will also consider the potential degradation in threaded areas and locations of crevices and stress concentration. The staff recommends that the licensee proposed reinspection plan should also consider the plant specific repair design requirements, and the extent and the results of the baseline inspection, performed during pre-modification inspection. The staff will review the licensee's reinspection plans for the core shroud and repair assemblies when submitted. However, the licensee should submit their reinspection plans within 6 months after restart from the current refueling outage. Since the core shroud and its repair assemblies are classified as ASME Code Class B-N-2 components (core structural support), the reinspection plans when approved by NRC should be incorporated into the ASME Section XI inservice inspection (ISI) program.

The NRC staff also recommends that the licensee should incorporate the following when performing reinspection during the next refueling outage: (1) the qualification of the UT techniques should include a mock-up block which simulates the configuration of the H8 dissimilar metal weld; and (2) the development of an effective method to locate the segment welds of the top guide support ring.

The NRC staff has reviewed the licensee's inspection results. The staff concludes that the licensee's inspection is acceptable to support the planned core shroud repair. Although some cracks were found, they are minor and would not impact the structural integrity of the welds during the operation in the next fuel cycle.

3.0 CONCLUSION

The proposed core shroud repair has been designed as an alternative to the requirements of ASME Code pursuant to Title 10, Code of Federal Regulations, Part 50.55a(a)(3)(1). Based on a review of the shroud modification hardware from structural, systems, materials, and fabrication considerations, as discussed above, the staff concludes that the proposed modifications of the Nine Mile Point 1 core shroud are acceptable.

Principal Contributors: J. Rajan, K. Kavanagh, W. Koo

Dated: March 31, 1995

NIAGARA MOHAWK

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CARL D. TERRY
Vice President
Nuclear Engineering

March 27, 1995
NMP1L 0932

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Subject: *Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors" (TAC No. M90102)*

Gentlemen:

During installation of the core shroud repair at Nine Mile Point Unit 1 (NMP1), Niagara Mohawk conducted inspections of the core shroud horizontal welds H2 and H3. The inspections of the H2 and H3 welds were performed with the intent to determine which assumptions of analyzed Cases 1, 2 or 3, (GENE-B13-01739-04 Supplement 1) are valid. The inspections were performed in accordance with the visual inspection requirements provided by the BWRVIP document "Core Shroud NDE Uncertainty and Procedure Standard" dated November 21, 1994. That document details the requirements for enhanced visual examination (EVT) of core shroud welds being examined for Intergranular Stress Corrosion Cracking (IGSCC), including such attributes as pre-cleaning, lighting, camera resolution and inspector training. The inspection findings are summarized below. The examination data sheets are provided in Enclosure 1 to this letter. A sketch of the NMP1 shroud welds is included as Enclosure 2

Shroud weld H2 was inspected at four locations adjacent to the repair tie rod locations on the outside shroud surface. Each inspected location was approximately 36 inches in length and included both upper and lower weld heat affected zones (HAZs). The upper weld HAZ is in the shroud shell plate material. The lower weld HAZ is in the top guide support ring outer vertical surface. The inspection identified one 2 inch long indication on the weld lower HAZ at the 90 degree azimuth, which was evaluated by the examiner as acceptable. The indication is a non-relevant surface mark. All other areas inspected in both the upper and lower HAZ were acceptable and were without indications of flaws.

Shroud weld H3 was inspected at four locations adjacent to the repair tie rod locations on the inside shroud surface. Each inspected location was approximately 36 inches in length and included both upper and lower weld HAZs. The upper weld HAZ is in the top guide support ring inner vertical surface. The lower weld HAZ is in the shroud shell plate material. All areas inspected in the lower HAZ were acceptable and were without indications or flaws. The inspection identified rejectable indications, characterized as IGSCC, at three of the four

H-3

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locations in the weld upper HAZ. The cracking extended through essentially all of the 36 inches at each of the three locations. The fourth location in the weld upper HAZ, at azimuth 350 degrees, was acceptable and was without indications or flaws.

In conclusion, the indications found are consistent with industry inspection findings to date. IGSCC in the weld HAZ of plate ring 304 material at similar locations is not unexpected and is understood. Niagara Mohawk believes that the inspections are a representative sample of the uninspected portions of the H2 and H3 welds. The inspection results of the HAZ on the ring and shell sides of the H2 and H3 welds supports the assumptions used in analysis Case 1. The analysis Case 2 assumed cracks on the shell side of both welds. The inspection found no evidence of cracks in these locations. The analysis Case 3 assumed simultaneous cracks on both shell and ring side. Evidence of this combination of cracks was not found.

NMPC has concluded therefore that analysis Cases 2 and 3 [illegible] by the inspection and that these cracking scenarios are not realistic for NMP[illegible].

Very truly yours,

C. D. Terry
Vice President - Nuclear Engineering

CDT/JMT/lmc

xc: Regional Administrator, Region I
Mr. L. B. Marsh, Director, Project Directorate [illegible]
Mr. G. E. Edison, Senior Project Manager, NRR
Mr. B. S. Norris, Senior Resident Inspector
Records Management

ENCLOSURE 1

1-2.01-95-0261

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GE Nuclear Energy

INVESSEL VISUAL EXAMINATION DATA SHEET

SITE: Nine Mile UNIT: 1 PROJECT NUMBER: 1ETED	CAMERA RESOLUTION	TYPE OF VISUAL EXAMINATION
	☐ .001" Diameter Wire ■ .0005" Diameter Wire ☐ 1/32" Black Line	☐ VT-1 ☐ DIRECT ☐ VT-3 ■ REMOTE ■ ENHANCED VT-1

Equipment Used During the Examination	
BTV-1250 Underwater Camera with twin 50's Super VHS Recorder	Procedure No. VT-NMP-201V0 Revision No. 0 FRR No. NMP-015

Component / Area Viewed	Description of Recordable Indications	Ind. No.	Accept	Reject	Film Footage
1/2 Mil Camera Resolution			X		0:00:00
Shroud Weld H-3 90° I.D.	36" Indication Top Side of Weld, No Indications Bottom of Weld			X	0:00:54
Shroud Weld H-3 170° I.D.	36" Indication Top Side of Weld, No Indications Bottom of Weld			X	0:24:09
Shroud Weld H-3 350° I.D.	No Indications Observed. 36" Inspected		X		0:38:51
Shroud Weld H-3 270° I.D.	36" Indication Top Side of Weld, No Indications Bottom of Weld			X	0:52:19
1/2 Mil Camera Resolution			X		1:28:47
Shroud Weld H-2 90° O.D.	2" Long Ind. Bottom of Weld Tape Count [illegible] 36" Inspected		X		1:29:57

Examined by:	Level
Wayne Farrell	II
Kirk Robideau	II
Michael Sharlow	II

Date Performed: March 13, 1995

COMMENTS

REFER TO [illegible] 95-0807 FOR REJECTS @ H-3

[signature] III 3-17-95
Reviewed Level Date

[signature] III 3/17/95
Reviewed Level Date TS 3/17/95

Report Number 9514
Tape Number 95-14

A

[signature] ANII 3-17-95

1-2.01.95-0261

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GE Nuclear Energy

INVESSEL VISUAL EXAMINATION DATA SHEET

SITE: Nine Mile UNIT: 1
PROJECT NUMBER: 1ETED

CAMERA RESOLUTION
- ☐ .001" Diameter Wire
- ■ .0005" Diameter Wire
- ☐ 1/32" Black Line

TYPE OF VISUAL EXAMINATION
- ☐ VT-1
- ☐ VT-3
- ■ ENHANCED VT-1
- ☐ DIRECT
- ■ REMOTE

Equipment Used During the Examination
ETV-1250 Underwater Camera with twin 50's
Super VHS Recorder

Procedure No. VT-NMP-201V0
Revision No. 0
FRR No. NMP-006

Component / Area Viewed	Description of Recordable Indications	Ind. No.	Accept	Reject	Film Footage
Camera Resolution 1/2 Mil			X		0:00:00
Shroud H-2 O.D. at 350°	36" Inspected		X		0:01:16
Shroud H-2 O.D. at 270°	36" Inspected		X		0:10:46
Shroud H-2 O.D. at 170°	36" Inspected		X		0:24:22
Camera Resolution 1/2 Mil			X		0:34:47

Examined by: Level
Kirk Robideau II
Michael Sharlow II

Date Performed: March 13, 1995

COMMENTS
NOTE: APP 3 " OF EACH AREA WAS EXAMINED.

[illegible] 3-15-95
Reviewed Level Date

Ed [illegible] 3/17/95
Reviewed Level Date

Report Number 9515
Tape Number 95-15

A

[illegible] 3-17-95

ENCLOSURE 2

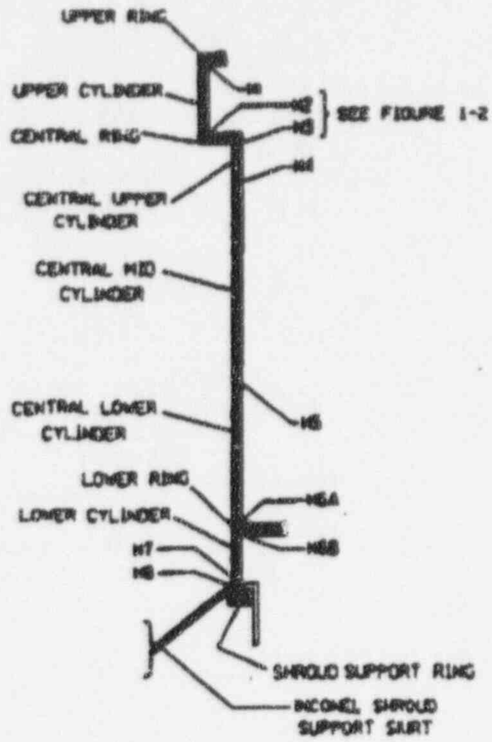

FIGURE 1-1 NINE MILE POINT 1 SHROUD WELDS

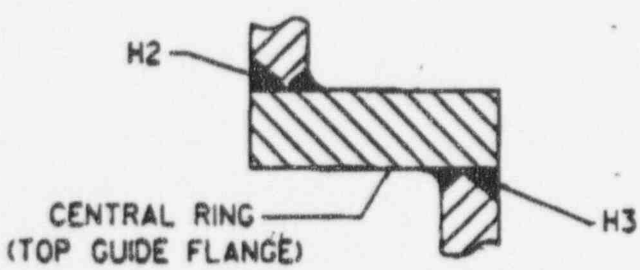

FIGURE 1-2 DETAIL TOP GUIDE FLANGE