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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING RESPONSE TO GL 94-03 FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

1.0 INTRODUCTION

The core shroud in a Boiling Water Reactor (BWR) is a stainless steel cylindrical component within the reactor pressure vessel (RPV) that surrounds the reactor core. The core shroud serves as a partition between feedwater in the reactor vessel's downcomer annulus region and the cooling water flowing up through the reactor core. In addition, the core shroud provides a refloodable volume for safe shutdown cooling and laterally supports the fuel assemblies to maintain control rod insertion geometry during operational transients and accidents.

In 1990, crack indications were observed at core shroud welds located in the beltline region of an overseas BWR. This reactor had completed approximately 190 months of power operation before discovery of the cracks. As a result of this discovery, General Electric Company (GE), the reactor vendor, issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications," on October 3, 1990, to all owners of GE BWRs. The RICSIL summarized the cracking found in the overseas reactor and recommended that at the next refueling outage, plants with high-carbon Type 304 stainless steel shrouds perform a visual examination of the accessible areas of the seam welds and associated heat-affected zone (HAZ) on the inside and outside surfaces of the shroud.

Subsequently, a number of domestic BWR licensees 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 inspection experience. Of the inspections performed to date, significant cracking was reported at several plants. The combined industry experience from these plants indicates that both axial and circumferential cracking can occur in the core shrouds of GE designed BWRs.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03 to all BWR licensees (with the exception of Big Rock Point) to address the potential for cracking in their core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

Enclosure

- inspect their 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; and

- work closely with the Boiling Water Reactor Owners Group (BWROG) on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking.

Niagara Mohawk Power Corporation (NMPC), the licensee for Nine Mile Point Nuclear Station Unit No. 1 (NMP-1), responded to GL 94-03 on August 23, 1994. Part of the licensee's response included a safety assessment to justify continued operation of NMP-1 until the inspection of core shroud scheduled for February 1995 refueling outage. NMPC stated that they will provide the NRC with their plans for inspection, evaluation and/or repair based on BWR Vessel and Internals Project (BWRVIP) recommendations no later than three months prior to performing the inspection, and submit inspection results within 30 days from the completion of the inspection. The NRC staff requested additional information on NMPC's response to GL 94-03. On October 14, 1994, NMPC presented additional information to the staff at NRC Headquarters regarding the structural integrity assessment of NMP-1 weld H8. During the meeting, NMPC also showed the NRC staff portions of their video-tape recording of previous inspections of weld H8.

2.0 JUSTIFICATION FOR CONTINUED OPERATION AND SCHEDULE FOR INSPECTION/REPAIR

NMPC will inspect or repair the NMP-1 core shroud, as appropriate, during the next refueling outage scheduled for February 1995. The following is the staff's assessment of the licensee's basis for justifying continued operation of NMP-1.

2.1 Susceptibility of the NMP-1 Core Shroud to IGSCC

The core shroud cracks which are the subject of GL 94-03, result from intergranular stress corrosion cracking (IGSCC) which is most often associated with sensitized material near the component welds. IGSCC is a time-dependent phenomena requiring a suceptible material, a corrosive environment, and a tensile stress within the material.

Industry experience has shown that austenitic stainless steels with low carbon content are less susceptible to IGSCC than stainless steels with higher carbon content. The formation of carbides at the grain boundaries upon moderate heating (sensitization) is hindered for Type 304 stainless steels with carbon contents below 0.03%. BWR core shrouds are constructed from either Type 304 or Type 304L stainless steel. The slightly lower carbon content of Type 304L (< 0.035%) makes it less prone to develop IGSCC.

Currently available inspection data indicate that core shrouds fabricated with forged ring segments are more resistant to IGSCC than rings constructed from welded plate sections. The current understanding for this difference is related to the surface condition resulting from the two shroud fabrication processes. Welded shroud rings are constructed by welding together arcs machined from rolled plate. This process exposes the short transverse direction in the material to the reactor coolant. Elongated grains and stringers in the material exposed to the reactor coolant environment are believed to accelerate the intiation of IGSCC.

Water chemistry also plays an important role in regard to IGSCC susceptibility. Industry experience has shown that plants which have operated with a history of high reactor coolant conductivity have been more susceptible to IGSCC than plants which have operated with lower conductivities[1]. Furthermore, industry experience has shown that reactor coolant systems (RCSs) which have been operated at highly positive, electro-chemical potentials (ECPs) have been more susceptible to IGSCC than RCSs that have been operated at more negative ECPs[2]. The industry has made a considerable effort to improve water chemistry at nuclear facilities over the past ten years. Industry initiatives have included the introduction of hydrogen water chemistry as a means of lowering ECPs (i.e., making the ECPs more negative) in the RCS. The effectiveness of hydrogen water chemistry in reducing the susceptibility of core shrouds to IGSCC initiation has not been fully evaluated; however, its effectiveness in reducing IGSCC in recirculation system piping has been demonstrated.

Welding processes can introduce high residual stresses in the material at the weld joint. The high stresses result from thermal contraction of the weld

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[1]Conductivity is a measure of the anionic and cationic content of liquids. As a reference, the conductivity of pure water is ~0.05 μS/cm. Reactor coolants with conductivities below 0.20 uS/cm are considered to be relatively ion free; reactor coolants with conductivities above 0.30 μS/cm are considered to have a relatively high ion content.

[2]The electrochemical potential (ECP) is a measure of a material's susceptibility to corrosion. In the absence of an externally applied current, and, therefore, for reactor internals in the RCS, the electrochemical potential is equal to the open circuit potential of the material. Industry experience has shown that crack growth rates in reactor internals are low when the ECP $\leq$ ~-0.230 volts.

metal during cooling. A higher residual tensile weld stress will increase the material's susceptibility to IGSCC. Although weld stresses are not easily quantified, previous investigation into weld stresses indicate that tensile stresses on the weld surface may be as high as the yield stress of the material. The stress decreases to compressive levels in the center of the welded section.

NMPC has reviewed the materials, fabrication and operational histories (water chemistry and on-line years) of the NMP-1 core shroud and has submitted this information to the staff in its response to GL 94-03.

The core shroud of NMP-1 is susceptible to IGSCC. The NMP-1 susceptibility ranking is very high among all domestic operating BWRs. NMP-1 is considered to have a high potential of 360° cracking in the core shroud. Some of the plants (Brunswick Unit 1, Dresden Unit 3 and Quad Cities Unit 1) with similar susceptibility rankings have detected extensive 360° cracking in the core shroud. The NMP-1 plant-specific susceptibility factors are summarized below:

(i) There are four support rings in the NMP-1 core shroud. With the exception of the shroud support ring, all other rings (upper ring, central ring and lower ring) were fabricated by welding two rolled plate segments, followed by machining to size. The shroud support ring was made of forging, and was sensitized during the initial post-weld heat treatment of the vessel. The shroud cylinders and all its support rings were made of Type 304 stainless steel materials with the carbon content varying from 0.042% to 0.064%.

(ii) All shop welds (H1 to H6) of the core shroud were fabricated by submerged arc welding process using ER-308 filler metal. The residual stresses resulting from this welding process are expected to be high.

(iii) NMP-1 had high average coolant conductivity for the first five cycles (0.456 μS/cm).

(iv) NMP-1 has operated more than 14.4 on-line years.

Considering the above plant-specific susceptibility factors as well as the industry-wide inspection experiences and the uncertainties in the residual stress profile resulting from fabrication, the staff concludes that significant cracking in the NMP-1 core shroud can not be ruled out.

2.2 Basis for Continued Operation

2.2.1 Structural Integrity Assessment of Core Shroud Welds

(a) Core Shroud Welds H1 Through H7

NMPC stated that the structural margins for welds H1 through H7 at NMP-1 is bounded by the conclusion of the generic assessment performed by

General Electric for BWROG (GENE-523-A107P-0794, Revision 1, "BWR Shroud Cracking Generic Safety Assessment," August 1994.) that finding a 360 degrees through-wall crack with an average depth exceeding the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) allowable during the spring 1995 inspection is unlikely. However, because of the uncertainties associated with the residual stress profile assumed in the assessment, it would be necessary to use the 1994 fall inspection results of Oyster Creek to support this conclusion.

Since both Oyster Creek and NMP-1 units were fabricated by the same vendor (P. F. Avery) during the same time period, their susceptibility factors attributed from materials, fabrication and on-line operation years would be similar. Furthermore, NMP-1's water chemistry is bounded by that of Oyster Creek because NMP-1 has lower average conductivity (0.457 μS/cm) in the first five cycles than that of Oyster Creek (0.526 μS/cm). Therefore, the propensity of core shroud cracking at NMP-1 is expected to be bounded by that of Oyster Creek. The core shroud inspection results of Oyster Creek performed during the fall 1994 outage as discussed in Section 2.4 support the BWROG's generic conclusion.

(b) Core Shroud Weld H8

Weld H8 is considered the most limiting weld in the core shroud assembly because it provides the vertical support for the core shroud. Weld H8, connecting the stainless steel shroud support ring (Type 304 forging) to the Inconel 600 support cone, was fabricated with Inconel 182 filler metal. The shroud support assembly consisting of the support ring, support cone, and weld H8 was post weld heat treated (PWHT) at 1150 °F during vessel fabrication. The licensee performed a finite element linear elastic stress analysis using the ANSYS code to investigate the stress state of the shroud support assembly during normal operation. The stresses in the radial direction are the main focus in the stress analysis because they are the most safety limiting stresses, since the radial stresses would promote the cracking of weld H8 in a direction that could potentially lead to a loss of the capability of vertically supporting the core shroud. The stresses discussed below refer to the stresses in the radial direction. The results of the licensee's stress analysis showed that, after PWHT, the tensile weld residual stresses were reduced to 7 ksi; the highest tensile stresses, about 13 ksi, consisting of the residual stresses and normal operating stresses, are located at the top of weld H8 in the ring; and the stresses in the areas adjacent to weld H8 are mostly compressive. Therefore, NMPC concluded that crack initiation in such a low stress field is not likely and even if cracks are initiated, the crack growth in the areas adjacent to the weld is expected to be very small due to the presence of compressive residual stresses. NMPC also reported that only 1/4 inch of the ligament is needed in weld H8 to support the LOCA loads resulting from the recirculation pipe break event. NMPC also performed an elastic-plastic finite element analysis. In this analysis the average weld radial shrinkage is assumed to be about 0.25 inch. This shrinkage would

yield a net average radial tension stress of about 1 ksi. Due to the presence of this 1 ksi radial tension stress, deep cracks along the weld interface would expected to have a top surface opening of about 0.194 inch, which would make the cracks easily detectable during visual examination. Assuming an initial fabrication flaw size of 0.01 inch (limiting flaw size after liquid penetrant inspection) NMPC's crack growth calculation has shown that there is adequate remaining ligament at weld H8 at the end of cycle 11 (current cycle) to support normal operating and LOCA loads. Based on the results of the linear elastic stress analysis and elastic-plastic fracture mechanics analysis, NMPC concluded that the structural integrity of weld H8 would be maintained during operation until the next refueling outage scheduled for February 1995.

The licensee did not provide supporting test data to validate quantitatively the assumed residual stress through-wall profile. Although the staff considers that sufficient ligament will remain at H8 to ensure adequate structural integrity until the next inspection, the crack growth in weld H8 could be much more significant if a more aggressive tensile residual stress profile is considered in the analysis. The licensee should demonstrate by testing that the residual stress profile assumed in their assessment model is conservative.

2.2.2 Previous Core Shroud Inspection

NMPC performed in-service visual inspection of welds H7 and H8 at 100% accessible weld areas (about 568 inches on the outside surface) in 1986, 1988, and 1993. Weld H7 is a field weld which connects the lower cylinder to the sensitized shroud support ring. Weld H8 is a bimetallic weld (Inconel 182 weld metal) connecting the Inconel 600 support cone to the sensitized shroud support ring (Type 304 stainless steel forging). These visual examinations are considered meeting the GE's SIL 0572, Revision 1, guidelines with the exception that surface cleaning was not performed prior to inspection. During a 1989 vessel annulus access study, a small portion of welds H1 through H4 was observed. No indications were found from any of those inspections.

Based on the industry-wide inspection experiences, the surface cleaning is an important practice for a good and reliable visual inspection of the core shroud for the detection of tight IGSCC cracks. BWRVIP issued standards for visual inspection of core shrouds on September 8, 1994. The standards recommend that prior to inspection, the surface should be free from oxide deposits or other conditions that would prevent the detection of indications. The subject inspections do not meet the BWRVIP standards. However, the industry experiences have also shown that deep and significant cracking could be detected without surface cleaning such as the cracking of H3 weld at Brunswick Unit 1. At NMP-1, the through wall cracking of control rod drive stub tubes was visually observed without surface cleaning. The NRC staff reviewed portions of the video-tape recording of the visual inspections of weld H8 and has determined that the quality of the visual inspections is

acceptable for the purpose of detecting deep, significant cracking. Therefore, based on the results of the previous inspections and the industry inspection experiences, the NRC staff believes that even if there are cracks at welds H7 and H8, the cracking would not be extensive and deep.

2.2.3 Application Of Oyster Creek Core Shroud Inspection Results

During the fall 1994 refueling outage, General Public Utilities (GPU) Nuclear inspected the core shroud at Oyster Creek. For welds H1 through H6 the inspection was performed mainly by means of ultrasonic examination, using GE's outside diameter tractor or suction cup delivery systems and GE SMART 2000 for data collection. The ultrasonic examination employed a number of transducers for flaw detection, which consisted of 45° shear, 60° and 80° longitudinal. The 80° longitudinal is used for generating creeping wave for the detection of surface and near surface flaws. Due to the limitation in accessibility, only visual inspection is performed on weld H9 and 30 brackets. The performance of visual inspection follows the BWRVIP guidelines. Inspection was not performed on welds H7 and H8 because Oyster Creek has 36 redundant brackets designed to replace the functions of welds H7 and H8. Minor cracking was found on H2, H3, H6A, and H6B welds. Weld H4 was reported to have the most significant cracking since cracks were detected in all inspected weld areas (about 49% of circumference). The preliminary sizing of the cracks at H4 indicated the cracks were not deep enough to compromise structural integrity with the shroud. Based on economic considerations, GPU Nuclear opted to repair the core shroud using MPR designed tie rod assemblies even though the safe operation of Oyster Creek could be justified by flaw evaluation for at least one more cycle without repair.

2.3 Licensee's Consequence Assessment of Shroud Response to Structural Loads

GL 94-03 requested that licensees perform a consequence assessment of the shroud response to design basis loads and their effect on the ability of plant safety features to perform their function assuming 360° through-wall cracking. NMPC's intent of this consequence assessment was to demonstrate that fuel geometry and core cooling would be maintained given the unlikely occurrence of a through-wall failure of any horizontal weld, and to identify whether horizontal weld failures would be detectable. Fuel geometry must be maintained to ensure control rod insertion while core cooling is ensured by proper emergency core cooling system (ECCS) performance. Based on the BWROG core shroud cracking generic safety assessment, NMPC concluded that weld separation during normal operations was detectable and provided off-normal operating procedures to the operators. NMPC, employing NMP-1 plant specific power, core flow, steam flow, and differential pressures across the shroud head and the shroud support as input to the Oyster Creek RELAP model, concluded that the ability to maintain reactivity control, fuel geometry, and core cooling was assured with substantial margin although degraded performance was assumed for design basis events. Based on this assessment, NMPC concluded that core shroud separation and/or displacement occurring during normal

operations or during anticipated events would have no effect on the primary safety functions of reactivity control and core cooling which are required to mitigate those events.

2.4 NRC Staff's Evaluation of Justification for Continued Operation

The NRC staff has reviewed the inspection results for other BWRs and notes that there has been no instance where a 360° through-wall crack existed in any plant that was inspected. Further, no BWR has exhibited any symptoms (power to flow mismatch) caused by leakage through a 360° through-wall crack. All analyses performed by licensees for plants with significant cracking in the core shroud showed that adequate ligaments would exist to assure structural integrity. In addition, there is a low probability for an initiating event which could potentially challenge the integrity of the core shroud, and particularly, since there is only a short duration of operation until the licensee implements necessary inspections or repairs.

Significant cracking in the NMP-1 core shroud cannot be ruled out based on susceptibility consideration. However, considering the results of NMP-1 previous core shroud inspections (Section 2.2.2) and the recent inspection performed at a plant (Oyster Creek, Section 2.2.3), which is more susceptible to IGSCC than NMP-1, the staff concludes that the NMP-1 core shroud should have sufficient ligament at the end of the proposed operating period to preclude failure under all conditions.

The NRC staff performed a qualitative assessment of the NMPC's consequence assessment. The NRC staff found the submittal to be a relatively complete assessment of the consequences of a main steam line break (MSLB), a recirculation line break (RLB) with acoustic and blowdown loads, a MSLB plus seismic event, and a RLB plus seismic event with regards to the unique features in the NMP-1 design. The NRC staff could not entirely verify all the details of the evaluations by NMPC for NMP-1 such as the uncertainties in the assumptions of break development time and core spray piping restraint used in the MSLB analysis, and the detectability of all postulated failed horizontal welds. For a MSLB, NMPC's calculations demonstrated that the top guide would not lift above the fuel, therefore assuring no lateral fuel movement. Due to the aforementioned uncertainties, the NRC staff determined that there is some likelihood of top guide lift above the fuel for upper weld locations. However, even if this were to occur, the NRC staff concluded that safe shutdown of the reactor should be achieved by the activation of the standby liquid control system (SLCS). Assuming the presence of through-wall failures of shroud welds, the other initiating event of concern would be the RLB. NMPC's calculations indicated momentary tipping of the shroud at certain postulated failed weld locations due to the blowdown forces, but no permanent lateral movement. For such shroud response, the staff agrees that adequate core flooding will be maintained since little core/annulus bypass will occur. Modeling the behavior of a cracked shroud during a RLB is quite complex, involving assumptions on crack surface friction and other interacting forces in the vertical and lateral directions. Therefore, the staff is unable to conclude with high confidence that such lateral motion cannot occur following

a RLB. Large lateral movement could damage the core spray lines and limit adequate post loss-of-coolant accident (LOCA) cooling. Although the NRC staff could not agree with NMPC's assessment of the MSLB with an assumed through-wall crack at the upper shroud welds and the RLB with an assumed through-wall crack at the lower shroud welds, the NRC staff concluded that only the most extreme assumptions, with respect to initiating events, crack locations, and crack depths, would result in unacceptable consequences. Therefore, due the low frequency of the initiating event, the availability of the SLCS, and the presence of remaining ligament to assure structural integrity, the NRC staff concludes that there is no undue risk to the public health and safety.

Furthermore, due to the unique design features of BWR 2s, NMPC stated that the integrity of the H8 weld is essential. The licensee has performed a finite element linear elastic stress analysis and inspections to demonstrate the integrity of the H8 weld. The NRC staff agrees that the integrity of the H8 weld must be ensured to justify power operation until the next scheduled refueling outage. Based on previous inspections performed at NMP-1, the NRC staff determined that extensive cracking at H8 is unlikely, and therefore, remaining ligament will assure structural integrity. Therefore, the NRC staff concluded that power operation is acceptable until the next scheduled refueling in February 1995.

3.0 CONCLUSIONS

Based on the evaluation provided in Section 2, the NRC staff finds that the schedule for the inspection or preemptive repair of the core shroud at NMP-1 is acceptable. The staff concludes that NMP-1 can continue to be safely operated until their next refueling outage.

4.0 OUTSTANDING ISSUES/FUTURE ACTIONS

In accordance with the reporting requirements of GL 94-03, NMPC shall submit to the NRC, no later than 3 months prior to performing the core shroud inspections, both the inspection plan and NMPC's plans for evaluating and/or repairing of the shroud based on the inspection results. In addition, results should be provided to the NRC within 30 days from the completion of the inspection. If NMPC identifies any core shroud cracking requiring an analysis per the ASME Code, details of such evaluations must also be submitted to the NRC for review.

It should be noted that the industry is currently encountering difficulty in performing comprehensive inspections of lower shroud welds due to NDE equipment accessibility problems. The staff urges licensees to work with various vendors and the EPRI NDE Center in order to develop improved reliable tooling for inspections of shroud welds which are highly obstructed. Should improved inspections techniques become available, the staff recommendation is for licensees to reinspect the lower shroud welds at the earliest opportunity.

NMPC indicated in their response that they may adjust the NMP-1 core shroud inspection schedule and scope per guidance from the BWRVIP. At present, the NRC has not approved the inspection guidelines proposed by the BWRVIP. Considerable differences remain with regard to the recommended scope of core shroud inspections. The NRC staff cautions NMPC against modifying their plans according to BWRVIP recommendations which have not undergone review and approval by the NRC. The staff's current position with regard to the scope of inspections is a recommendation for the inspection of 100% of the accessible core shroud welds. Should the licensee opt to install a preemptive repair in lieu of performing a comprehensive core shroud inspection, the only required inspection is that mandated in the staff approval of the repair option.

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Date: January 13, 1995

The staff concludes that based on an assessment of the IGSCC susceptibility factors pertaining to the core shroud material properties, fabrication, and the past operational history as well as the industry-wide inspection experiences, extensive cracking in the NMP-1 core shroud cannot be ruled out. However, the core shroud cracking at NMP-1 is expected to be bounded by that of Oyster Creek; since both units were fabricated by the same vendor during the same time period, and NMP-1 had better water chemistry than that of Oyster Creek in the early five fuel cycles. Oyster Creek recently completed its core shroud inspection of welds H1 through H7. Extensive cracking was found at weld H4, however, the results of preliminary sizing showed that the cracks were not sufficiently deep to compromise structural integrity with the shroud. Based on previous core shroud inspections at NMP-1 and the recent Oyster Creek inspection results, NMP-1 core shroud is not likely to contain cracks which would compromise the structural integrity of the core shroud during the remainder of the current fuel cycle. In addition, other BWRs with core shrouds considered more susceptible to IGSCC have not identified any 360° through-wall cracking during inspections. Per the reporting requirements of GL 94-03, NMPC has submitted (by letter dated November 11, 1994) a plan for inspecting the NMP-1 core shroud during the refueling outage scheduled to begin in February 1995. NRC staff review of this inspection plan is in progress. NMPC will be notified regarding our evaluation of this plan in separate correspondence.

Sincerely,

Original signed by:

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Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosure: Safety Evaluation

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