UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
NINE MILE POINT NUCLEAR STATION, UNIT 2
PROPOSED USE OF ALTERNATIVE EXAMINATION
FOR THREE MAIN STEAM LINE SAFETY RELIEF VALVES
NIAGARA MOHAWK POWER CORPORATION
DOCKET NO. 50-410

1.0 INTRODUCTION

Technical Specification (TS) 4.0.5a for Nine Mile Point Nuclear Station, Unit 2 (NMP2), states that inservice inspection (ISI) of Class 1, 2, and 3 components of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) shall be performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), "Inservice Inspection Requirements," except where specific written relief has been granted by the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 10-year interval, subject to any limitations and modifications listed in the regulations. The applicable edition of Section XI of the ASME Code for the NMP2 second 10-year ISI interval is the 1989 Edition. In a letter dated June 29, 1998, Niagara Mohawk Power Corporation (NMPC or licensee), requested that the NRC staff approve an alternative to the examination requirements of paragraph IWA-5250(a)(2) to Section XI of the ASME Code for the bolted connections of three Code Class 1 NMP2 main steam line safety relief valves designated as 2MSS*PSV125, 2MSS*PSV128, and 2MSS*PSV137 (see Figures 10.1-3a-d in the NMP2 Updated Safety Analysis Report).

2.0 BACKGROUND

The licensee submitted the request to use an alternative to the requirements of ASME Code, Section XI, paragraph IWA-5250(a)(2). The Code of record for the second interval is the ASME Code, Section XI, 1989 Edition. The information provided by the licensee in support of the request has been evaluated and the bases for disposition are documented below:

2.1 Code Requirement

ASME Code, Section XI, paragraph IWA-5250(a)(2) states: "If leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100."

2.2 Licensee's Code Relief Request

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief from the requirement to remove the bolts and VT-3 visually examine the bolts after leakage was detected at the flanges of three main steam line safety valves during the reactor pressure vessel (RPV) system leakage test performed in June 1998 during RFO6.

2.3 Licensee's Basis for Requesting Relief

In its letter dated June 29, 1998, the licensee states:

"NMPC is proposing an alternative in accordance with 10 CFR 50.55a(a)(3), on the basis that the proposed alternative provides an acceptable level of quality and safety, and that compliance with the Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

During the RF05 RPV system leakage test (performed in October, 1996, under the 1983 Code), these three SRVs [safety relief valves] exhibited a small amount of leakage at the pressure retaining inlet flange. During the subsequent startup, these three SRVs were re-inspected at approximately 900 psi saturation conditions, and there was no leakage. The RF06 RPV system leakage test was the first RPV system leakage test performed during the second ten-year interval, which began on April 5, 1998. The three subject SRVs were installed with new bolting during RF05, which ended in November 1996. These three SRVs have again exhibited a small amount of leakage during the RF06 RPV system leakage test.

The present bolting has been in service during the period that began with the startup that ended RF05 and ended with the shutdown that began RF06 (approximately 1½ years of service). Prior to installation, the flange surfaces were inspected for corrosion and damage during the performance of the mechanical maintenance procedure that tests and refurbishes the SRVs.

As documented on individual NIS-2 (Owner's Report for Repairs or Replacements) forms, the bolting on the three subject SRVs was replaced in 1996. This replacement was for the convenience of maintenance, and not due to service-induced failure. The replacement bolting was procured by NMPC purchase order and inspected to the requirements of a procurement requirements evaluation form (PREF). Samples of the batch were subjected to destructive examination, and every part was examined by either magnetic particle or dye penetrant. The manufacturer certified that the requirements of ASME Section III and the PREF were met, and NMPC performed a source surveillance to provide additional assurance that the conditions of the Code and the PREF were fulfilled.

The subject SRVs had a small observable leakage during the RF05 RPV system leakage test in October, 1996. During the subsequent startup, this leakage had stopped by the time the plant reached saturation conditions at approximately 900 psi. This further demonstrates that the leakage was not due to a problem with the pressure-retaining bolting.

Removing the bolting on an SRV requires removal of the SRV. The removal and installation of these three SRVs results in an estimated occupational exposure of approximately 3.5 person-rem. This dose is not justified, since removal and inspection of the bolting does not provide any assurance of improved quality in the bolted connection.

In summary, the basis for the alternative examinations proposed below is that:

1. The small leakage observed during the RF05 system leakage test self-corrected when the piping system was heated at saturated steam conditions. It is expected that this self correction will occur again during the heatup following RF06.

2. The small leakage observed during the RPV system leakage test is not indicative of corrosion or other degradation of the bolted connection.

3. Removal and inspection of the bolting provides no additional assurance or improvement in the quality of the bolted connection.

4. The occupational exposure of 3.5 person-rem is not justified by a commensurate increase in safety, quality, or reliability."

2.4 <u>Licensee's Proposed Alternative Examination</u>

In its letter dated June 29, 1998, the licensee states:

"The alternative examinations apply only to the requirements for corrective measures specified in IWA-5250(a)(2). The performance of the RPV system leakage test and the acceptance criteria for that test are unaffected by this alternative.

NMPC will perform the following alternative examinations for the three SRVs identified above that exhibit detectable leakage at the pressure-retaining flange:

1. A VT-3 visual examination shall be performed in accordance with the Inservice Pressure Testing (ISPT) Acceptance Criteria on the visible portions of the flange and its bolting on each affected SRV.

2. During the startup following the RPV system leakage test, each of the three subject SRVs that exhibited detectable leakage during the RPV system leakage test shall be VT-2 examined with the reactor coolant system at saturation temperature and pressure ≥900 psig to confirm that the bolted connection is acceptable."

3.0 <u>EVALUATION</u>

The licensee has requested relief from the ASME Code requirements to remove bolting and perform VT-3 visual examination on three main steam line safety relief valves after leakage was detected during the RPV system leakage test during RFO6. The licensee stated that during the RF05 RPV system leakage test (performed in October 1996, under the 1983 ASME Code which did not require removal of the bolting), these three SRVs exhibited a small amount of leakage at the pressure retaining inlet flange. During the subsequent startup, these three SRVs were re-inspected at approximately 900 psi saturation conditions, and there was no leakage. The RF06 RPV system

leakage test was the first RPV system leakage test performed during the second 10-year interval, which began on April 5, 1998. The three SRVs were installed with new bolting during RF05, which ended in November 1996. These three SRVs have again exhibited a small amount of leakage during the RF06 RPV system leakage test. The licensee is following the same procedure for the valves as the one that was followed during the RFO5 outage because it appears that the valve flanges seal after saturation conditions are achieved at approximately 900 psig. The licensee will perform VT-3 visual examination in accordance with the acceptance criteria of the ASME Code, Section XI. Further, after saturation temperature and pressure of approximately 900 psig are reached during plant heatup, the flanges that exhibited leaks will be VT-2 examined to confirm that the bolting connections do not leak. This will ensure that the integrity of the bolted connections is maintained and that the joints do not leak in service.

4.0 CONCLUSION

The NRC staff concludes that the licensee has provided an acceptable alternative to the examination requirements of paragraph IWA-5250(a)(2) to Section XI of the ASME Code, inasmuch as the licensee's alternative program, if authorized, would provide an acceptable level of quality and safety.

Accordingly, pursuant to 10 CFR 50.55a(a)(3)(i), the alternative is authorized with respect to safety relief valves 2MSS*PSV125, 2MSS*PSV128, and 2MSS*PSV137 for use during RFO6 and the plant heatup that immediately follows RFO6.

Principal Contributors: G. Georgiev
D. Hood

Date: July 23, 1998

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
NINE MILE POINT NUCLEAR STATION, UNIT 2
PROPOSED USE OF ALTERNATIVE EXAMINATION
FOR ONE LOCAL POWER RANGE MONITOR
NIAGARA MOHAWK POWER CORPORATION
DOCKET NO. 50-410

1.0 INTRODUCTION

Technical Specification (TS) 4.0.5a for Nine Mile Point Nuclear Station, Unit 2 (NMP2), states that inservice inspection (ISI) of Class 1, 2, and 3 components of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) shall be performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), "Inservice Inspection Requirements," except where specific written relief has been granted by the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 10-year interval, subject to any limitations and modifications listed in the regulations. The applicable edition of Section XI of the ASME Code for the NMP2 second 10-year ISI interval is the 1989 Edition. In a letter dated June 29, 1998, Niagara Mohawk Power Corporation (NMPC or licensee), requested that the NRC staff approve an alternative to the examination requirements of paragraph IWA-5250(a)(2) to Section XI of the ASME Code for the bolted connection for one of the Code class 1 local power range monitors, designated as LPRM 56-33.

The NMP2 neutron monitoring system, which includes LPRMs, is discussed in Section 7.2.1.2.1 of the Updated Safety Analysis Report (USAR). The LPRMs are illustrated in USAR Figure 7.1-1, "Reactor Protection System," sheet 3.

2.0 BACKGROUND

The licensee submitted the request to use an alternative to the requirements of ASME Code, Section XI, paragraph IWA-5250(a)(2). The Code of record for the second interval is the ASME Code, Section XI, 1989 Edition. The information provided by the licensee in support of the request has been evaluated and the bases for disposition are documented below:

Enclosure 2

2.1 Code Requirement

ASME Code, Section XI, paragraph IWA-5250(a)(2) states: "If leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100."

2.2 Licensee's Code Relief Request

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief from the requirement to remove the bolts and VT-3 visually examine the bolts after leakage was detected at the flange of LPRM 56-33 during the reactor pressure vessel (RPV) system leakage test performed in June 1998 during RFO6.

2.3 Licensee's Basis for Requesting Relief

In its letter dated June 29, 1998, the licensee states:

"NMPC is proposing an alternative in accordance with 10 CFR 50.55a(a)(3), on the basis that the proposed alternative provides an acceptable level of quality and safety, and that compliance with the Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

Nine Mile Point Unit Two has 43 local power range monitor (LPRM) assemblies. Each LPRM assembly bolts to the bottom of the reactor pressure vessel (RPV) using a flanged connection. The flange has four bolts (actually hex-head cap screws) that secure the LPRM flange to the RPV flange. The RPV bottom head penetration for an LPRM assembly is approximately 1 3/8 inches in diameter.

The subject flange exhibited a very small detectable leakage (6 drops per minute) during the RPV system leakage test. During RFO5, four LPRMs (not 56-33) exhibited this small leakage during the RPV system leakage test and zero leakage during the 900 psi inspection performed during the subsequent startup that completed RFO5. The RFO6 RPV system leakage test is the first RPV system leakage test performed during the second ten-year interval, which began April 5, 1998. The four LPRMs that exhibited small detectable leakage during the RF05 RPV system leakage test did not exhibit any detectable leakage during the RF06 system leakage test

The bolt nearest the source of the leakage on LPRM 56-33 was removed and VT-3 inspected on June 26, 1998; an adjacent bolt was removed and VT-3 inspected on June 28, 1998. Neither bolt showed evidence of corrosion, damage, or degradation. The two inspected bolts were re-torqued after they were re-installed. Also on June 28, 1998, the bolt opposite the source of the leakage was scheduled to be removed and VT-3 inspected. When this third bolt was loosened, an unacceptable leakage from the flange resulted. The bolt was re-tightened to its required torque, and the leakage stopped. The fourth bolt was verified at that time to be torqued to the required value.

To remove the remaining two bolts from the LPRM 56-33 flange requires inserting a plug into the LPRM penetration from the inside of the reactor vessel. Removal of the remaining two bolts in the LPRM 56-33 flange constitutes an undue hazard to personnel if performed without a plug in the LPRM tube. Furthermore, it is an undue hardship without a compensating increase in safety to disassemble the reactor vessel head, remove the steam separator and dryer and other internals, remove at least four fuel bundles, and install a plug to permit removing the two remaining LPRM flange bolts for VT-3 examination.

Removal of all bolting when leakage occurs at a bolted connection does not provide an increase in the level of quality or safety because evidence of degradation can be determined by removing the one bolt closest to the source of leakage. If the removed bolt has evidence of degradation, then all remaining bolts would be removed, VT-3 examined, and evaluated in accordance with IWA-3100. Immediate removal of all bolting would require significant additional time, resources, and radiation dose.

The NRC's Safety Evaluation of Code Case N-416-1 for Nine Mile Point Nuclear Station, Unit Nos. 1 and 2, dated October 18, 1994, states that, 'The corrective actions with respect to removal of bolts from leaking bolted connections has been relaxed in the 1992 Edition, but use of this change has been accepted by the staff in previous Safety Evaluations.'

In summary, the basis for the alternative examination proposed below is that:

1. A VT-3 examination was performed on the bolt nearest the source of the identified leakage and an adjacent bolt. No evidence of corrosion or degradation was found. This examination exceeds the requirements of the 1992 Code.

2. The subject LPRM flange is exposed to high purity demineralized water at high temperature and pressure. This is not a corrosive environment for the materials in the flange and its bolting.

3. Compliance with the 1989 Code requirement to remove and inspect the remaining two bolts in the subject LPRM flange would impose an undue hardship without a compensating increase in the level of safety.

4. In the highly unlikely event that an LPRM flange should completely fail during power operation, the resulting coolant loss through the 1 3/8 inch hole is bounded by the small-break LOCA analysis (USAR Section 6.3 3.7.6)."

2.4 Licensee's Proposed Alternative Examination

In its letter dated June 29, 1998, the licensee states:

"In lieu of ASME Section XI, IWA-5250(a)(2), 1989 Edition, the 1992 Edition of ASME Section XI, IWA-5250(a)(2) shall be used. For the subject LPRM flange, the bolt nearest the source of leakage and an adjacent bolt were removed and VT-3 examined. No evidence was found of corrosion or degradation.

During the startup following the RPV system leakage test, the subject LPRM (56-33) flange shall be VT-2 examined with the reactor coolant system at saturation temperature and pressure ≥900 psig to confirm that the bolted connection is acceptable."

3.0 EVALUATION

The licensee has requested relief from the ASME Code requirements to remove bolting and perform the VT-3 visual examination on LPRM 56-33 after leakage was detected during the RPV system leakage test. The subject flange exhibited a very small detectable leakage (6 drops per minute) during the RPV system leakage test. The licensee stated that during the previous RFO5, four LPRMs (not 56-33) exhibited this small leakage during the RPV system

leakage test and zero leakage during the 900 psi inspection performed during the subsequent startup that completed RFO5. The RFO6 RPV system leakage test is the first RPV system leakage test performed at NMP2 during the second 10-year interval, which began April 5, 1998. The four LPRMs that exhibited small detectable leakage during the RF05 RPV system leakage test did not exhibit any detectable leakage during the RF06 system leakage test. The bolt nearest the source of the leakage on LPRM 56-33 was removed and VT-3 inspected on June 26, 1998; an adjacent bolt was removed and VT-3 inspected on June 28, 1998. Neither bolt showed evidence of corrosion, damage, or degradation. The two inspected bolts were re-torqued after they were re-installed. Also on June 28, 1998, the bolt opposite the source of the leakage was scheduled to be removed and VT-3 inspected. When this third bolt was loosened, an unacceptable leakage from the flange resulted. The bolt was re-tightened to its required torque, and the leakage stopped. The fourth bolt was verified at that time to be torqued to the required value.

The licensee is following the procedure required by the 1992 Edition of the ASME Code, Section XI, paragraph IWA-5250(a)(2). The procedure allows removal of the bolt that is nearest the source leakage and visually examining it per ASME Code VT-3 rules. Further, after saturation temperature and pressure of approximately 900 psig is reached, the flange that exhibited the leak will be VT-2 examined to confirm that the bolting connections do not leak. This will ensure that the integrity of the bolted connection is maintained and that the joints do not leak in service.

4.0 CONCLUSION

The NRC staff concludes that the licensee has provided an acceptable alternative to the examination requirements of paragraph IWA-5250(a)(2) to Section XI of the ASME Code, inasmuch as the licensee's alternative program, if authorized, would provide an acceptable level of quality and safety.

Accordingly, pursuant to 10 CFR 50.55a(a)(3)(i), the alternative is authorized with respect to LPRM 56-33 for use during RFO6 and the plant heatup that immediately follows RFO6.

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