May 15, 1987

Docket No. 50-410

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Mr. C. V. Mangan
Senior Vice President
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Mangan:

The Commission has issued the enclosed Amendment No. 2 to Facility Operating License No. NPF-54 for the Nine Mile Point Nuclear Station Unit 2 (NMP-2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 11, 1987, as supplemented March 16, 18 and 31, 1987 and April 2, 3, 7, 23 and 28, 1987.

This amendment revises the Technical Specifications related to the main steam isolation valves (MSIVs). Specifically, the amendment revises the trip setpoint and allowable value for the MSIV closure in Table 2.2.1-1 and to change the valve designations in Tables 3.6.1.2-1 and 3.6.3-1. Also, items 1.a.(2), (3) and (4) in Attachment 1 to the License are deleted. These changes result from the changeout of ball valves with globe valves. License Condition 2.C.(14) is the subject of a separate amendment.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/S/

Joseph D. Neighbors, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 2 to NPF-54
2. Safety Evaluation

cc: w/enclosures
See next page

PDI-1	PDI-1	PDI-1	OGC-Beth
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5/1/87	5/7/87	5/15/87	5/12/87

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Mr. C. V. Mangan
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. NPF-54

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated March 11, 1987, as supplemented March 16, 18 and 31, 1987 and April 2, 3, 7, 23 and 28, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by deleting items 1.a.(2), (3) and (4) of Attachment 1 and by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-54 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 2, are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Acting Director
Project Directorate I-1
Division of Reactor Projects, I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance:
May 15, 1987

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

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 2 TO FACILITY OPERATING LICENSE NO. NPF-54

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-3	2-3
3/4 6-6	3/4 6-6
3/4 6-23	3/4 6-23

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Intermediate Range Monitor, - Nuetron Flux - High	$\leq$120/125 divisions of full scale	$\leq$122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux - Upscale, Setdown	$\leq$15% of RATED THERMAL POWER	$\leq$20% of RATED THERMAL POWER
b. Flow-Biased Simulated Thermal Power - Upscale		
1) Flow-Biased 2) High-Flow-Clamped	$\leq$0.66 $(W-\Delta W)^{(a)}$ + 51%, with a maximum of $\leq$113.5% of RATED THERMAL POWER	$\leq$0.66 $(W-\Delta W)^{(a)}$ + 54%, with maximum of $\leq$115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - Upscale	$\leq$118% of RATED THERMAL POWER	$\leq$120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$1037 psig	$\leq$1057 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$159.3 in. above instrument zero*	$\geq$157.8 in. above instrument zero
5. Main Steam Line Isolation Valve - Closure	$\leq$8% closed	$\leq$12% closed
6. Main Steam Line Radiation - High	$\leq$3.0 x full-power background	$\leq$3.6 x full-power background
7. Drywell Pressure - High	$\leq$1.68 psig	$\leq$1.88 psig

* See Bases Figure B3/4 3-1.

(a) The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. $\Delta W=0$ for two loop operation. $\Delta W=5\%$ for single loop operation.

TABLE 3.6.1.2-1

ALLOWABLE LEAK RATES THROUGH VALVES IN

POTENTIAL BYPASS LEAKAGE PATHS

LINE DESCRIPTION	VALVE MARK NO	TERMINATION REGION	PER VALVE* LEAK RATE, SCFH
4 Main Steam Lines	2MSS*AOV6A, B, C, D 2MSS*AOV7A, B, C, D	Turbine Bldg.	6.0
Main Steam Drain Line (Inboard)	2MSS*MOV111, 112	Turbine Bldg.	1.875
Main Steam Drain Line (Outboard)	2MSS*MOV208	Turbine Bldg.	0.625
4 Postaccident Sampling Lines	2CMS*SOV77A, B 2CMS*SOV74A, B 2CMS*SOV75A, B 2CMS*SOV76A, B	Radwaste Tunnel	0.2344
Drywell Equipment Drain Line	2DER*MOV119 2DER*MOV120	Radwaste Tunnel	1.25
Drywell Equipment Vent Line	2DER*MOV130 2DER*MOV131	Radwaste Tunnel	0.625
Drywell Floor Drain Line	2DFR*MOV120 2DFR*MOV121	Radwaste Tunnel	1.875
Drywell Floor Vent Line	2DFR*MOV139 2DFR*MOV140	Radwaste Tunnel	0.9375
RWCU Line	2WCS*MOV102 2WCS*MOV112	Turbine Bldg.	2.5
Feedwater Line	2FWS*AOV23A 2FWS*V12A 2FWS*AOV23B 2FWS*V12B	Turbine Bldg.	12.0
CPS Supply Line to Drywell	2CPS*AOV104 2CPS*AOV106	Standby Gas Trtmt. Area	4.38
CPS Supply Line to Drywell	2CPS*SOV120 2CPS*SOV122	Standby Gas Trtmt. Area	0.625
CPS Supply Line to Supp. Chamber	2CPS*AOV105 2CPS*AOV107	Standby Gas Trtmt. Area	3.75
CPS Supply Line to Supp. Chamber	2CPS*SOV119 2CPS*SOV121	Standby Gas Trtmt. Area	0.625

* Test conditions: air medium, 40 psig.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
A. Automatic				
2MSS*AOV6 A,B,C,D	Inside MSIV	1	Z,X,C,D,E,P,T,R,RM,AA	3 to 5
2MSS*AOV7 A,B,C,D	Outside MSIV	1	Z,X,C,D,E,P,T,R,RM,AA	3 to 5
2MSS*MOV208	MSL Drain Line Outside IV	1	Z,X,C,D,E,P,T,R,RM,AA	18
2MSS*MOV111	Main Steam Drain Line Inside IV	1	Z,X,C,D,E,P,T,R,RM,AA	60
2MSS*MOV112	Main Steam Drain Line Outside IV	1	Z,X,C,D,E,P,T,R,RM,AA	60
2RHS*MOV33 A,B	RHS Cont. Spray Outside IVs	*	RM and *	35
2RHS*MOV104	RHS Reactor Head Spray Outside IV	5	A,L,M,Z,RM,CC,DD	50
2RHS*MOV40 A,B	Shutdown Cooling Return Outside IVs	5	A,L,M,Z,RM,CC,DD	29
2RHS*MOV67 A,B	SDC Inboard IV Bypass Valves	5	A,L,M,Z,RM,CC,DD	18
2RHS*MOV112	SDC Supply Inside IV	5	A,L,M,Z,RM,CC,DD	29
2RHS*MOV113	SDC Supply Outside IV	5	A,L,M,Z,RM,CC,DD	29
2CSH*MOV111	CSH Test Return to Suppression Pool Outside IV	*	RM and *	60
2ICS*MOV164	RCIC Vacuum Breaker Outside IV	11	H & F, RM	18
2CCP*MOV94 A,B	CCP Supply to RCS Inside IVs	8	B,F,Z,RM	30
2CCP*MOV17 A,B	CCP Supply to RCS Outside IVs	8	B,F,Z,RM	30
2CCP*MOV16 A,B	CCP Return from RCS Pumps Inside IVs	8	B,F,Z,RM	30
2CCP*MOV15 A,B	CCP Return from RCS Pumps Outside IVs	8	B,F,Z,RM	30
2DFR*MOV120	DFR Drain Tank Line Outside IV	8	B,F,Z,RM	45
2DFR*MOV121	DFR Drain Tank Line Inside IV	8	B,F,Z,RM	45

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 2 TO FACILITY OPERATING LICENSE NO. NPF-54

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-410

INTRODUCTION

By letter dated March 11, 1987, as supplemented March 16, 18 and 31, 1987 and April 2, 3, 7, 23, and 28, 1987, Niagara Mohawk Power Corporation (the licensee) requested an amendment to Facility Operating License No. NPF-54 for the Nine Mile Point Nuclear Station Unit 2. The amendment would revise the Technical Specifications related to main steam isolation valves (MSIV). The changes involve the MSIV closure setpoint and the valves' designation numbers. A License Condition, Section 2.C.(14), relating to special conditions appropriate only to the ball valves which have been removed is being addressed in a separate amendment. This amendment also deletes Items 1.a.(2), (3), and (4) of Attachment 1 to the license which relate only to the valves which have been removed.

As part of this amendment request, the licensee on March 18, 1987 requested that a Leakage Control System (LCS) not be required. The staff evaluated this request and issued a draft Safety Evaluation Report on April 14, 1987 which provided the staff's basis for not requiring an LCS. That Safety Evaluation Report is included in this Safety Evaluation as Attachment 1 and supports our conclusion that an LCS is not required. It should be noted that subsequent changes concerning the MSIVs or containment bypass leakage, such as changing the MSIV allowable leak rate in the Technical Specifications, or excessive leakage may require a reevaluation of the need for an MSIV LCS.

BACKGROUND

The MSIV's perform several functions such as Primary Containment isolation and Reactor Coolant pressure boundary. Industry experience described in NUREG-1169, "Technical Findings Related to Generic Issue C-8, Boiling Water Reactor Main Steam Isolation Valve Leakage Treatment Methods," indicates that MSIV leakage has been a concern. Ball valves were installed with the expectation that leakage would be reduced. However, experience with the ball valves has shown that they have not functioned as well as anticipated. Delamination of the tungsten carbide coating causes wearing between the seat and the ball which results in increased valve leakage. Packing leakage has also been a problem. Therefore, NMP-2 MSIVs were modified.

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The modification included cutting out the existing eight main steam isolation valves and replacing them with wye-pattern globe valves. The wye-pattern globe valves will meet all the same design criteria that the original design required. For example, the design will meet safety-related seismic and environmental qualifications; and IEEE 279 requirements. The globe valves were purchased from General Electric which normally supplies the valves as part of the NSSS contract.

This evaluation addresses the overpressurization protection analysis, LOCA analyses, transient and accident analyses, and actuation control system resulting from the change of the valves.

EVALUATION

Overpressurization Protection

The worst case overpressurization transient, MSIV closure with high flux scram, was not affected since failure of the MSIV direct position scram was assumed in the analysis. Therefore, the proposed MSIV closure trip setpoint change in the Technical Specifications, from "≤6% closed" to "≤8% closed" and allowable value change from "≤7% closed" to "≤12% closed" in RPS instrumentation setpoints, have no impact on the overpressurization protection analysis. The "≤8% closed" setpoint corresponds to a "≤12% closed" allowable value. This difference provides a margin for drift of the instruments. The "≤12% closed" allowable value corresponds to a "≤15% closed" analytical value (or ≥85% open). The difference here is a margin of conservatism.

Loss of Coolant Accident (LOCA)

The change in MSIV closure characteristics, resulting from the installation of the wye-pattern globe valves, has a negligible effect on the ECCS performance analyses as shown in Table 1. The change to wye-pattern globe valves would cause less than 1 degree F increase in the peak clad temperature (PCT) for the most limiting large break and less than 2 degrees F increase for small breaks. Therefore, the acceptance criteria for emergency core cooling systems for light water nuclear power reactors as contained in 10 CFR 50.46 are satisfied with the globe valves in operation. The modeling of steam flow during MSIV closure remains unchanged from that described on page B-9 of NEDO 10329, "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," and has been previously found to be acceptable by the staff.

In addition to reanalyzing the worst case breaks, the licensee assessed the impact of the change on other postulated breaks. For a recirculation line, feedwater line, or ECCS line break, MSIV closure is conservatively assumed to occur on Low-Low-Low water level (Level 1). A scram would be expected to have already occurred on Low water level (Level 3). Thus, changing the MSIV position scram setpoint has no effect on the ECCS performance analyses for these breaks since it was not utilized in these analyses.

For a steamline break inside the containment, the scram will occur on high drywell pressure before MSIV closure occurs. The MSIV position scram setpoint is not used for the ECCS system response. For steamline break outside the containment, the analysis conservatively starts with the water level at the scram trigger point, Low water level (Level 3). Realistically, a scram is likely to occur earlier due to MSIV closure on high steamline flow, but the scram input due to MSIV closure has been conservatively omitted in the analysis. Thus, the analysis is unaffected by the MSIV position scram setpoint change.

Anticipated Operation Occurrences

The proposed change to the MSIV closure setpoint necessitated by the valve change has been evaluated with respect to the transient and accident analyses contained in the FSAR. Loss of air or nitrogen, manual closure of all MSIVs, pressure regulator controller failure, and other transients and accidents were considered for any significant effect on the margin of safety.

The impact of a delayed scram signal due to the new MSIV closure-trip switch setpoint on transients has been evaluated. The new setpoint corresponds to an analytical limit of "85% MSIV open" instead of the previous "90% MSIV open." Two transients which take credit for this scram function are the manual closure of all main steam isolation valves (direct scram event) and the pressure regulator controller failure (open event). Of the two events, the manual closure is more limiting. The transient results are more sensitive (limiting) to the difference in the allowable range of the Technical Specifications (3 to 5 sec.) speed of MSIV closure (which is not being changed by this Technical Specification change) than due to a small scram delay resulting from the setpoint change. The proposed change to the Main Steam Isolation Valve-Closure setpoint was evaluated by reanalyzing the manual closure of all main steam isolation valves transient and there was no change in the critical power ratio (CPR) operating limit.

Another event affected by the setpoint change is load rejection without turbine bypass. This event was also reanalyzed. The change in Minimum Critical Power Ratio (MCPR), as shown in Table 1, is insignificant (much less than 0.01).

The remaining existing FSAR transient analyses are based upon an analytical model that bounds the closure characteristics (flow area versus time) of either the ball or globe valves. The wye-pattern globe valves have a 10 psi higher pressure differential when full open than the ball valves, due to frictional flow losses. Sensitivity studies performed by GE based upon information from a number of plants have shown that the larger differential pressure across the steamline volume produces milder transient response. Larger steamline differential pressure has a dampening effect on the pressure wave following a closure of turbine stop or control valves. Thus, since the previous analyses are based upon a model which conservatively simulates the wye-pattern valve characteristics, there is no significant impact on the other pressurization transients due to the MSIV change.

Actuation Control System

The Protection System signals that provide the trips for the wye-pattern MSIVs are the same signals utilized in the ball valve design. The power supplies are the same non-Class 1E 120VAC supplied by UPS3A (Trip System A) and UPS3B (Trip System B). The design utilizes the same electrical protection assemblies (EPA), distribution panels and the same cables.

The fail-safe de-energize to operate logic function, used for the ball valves, remains with the wye-pattern valves. This logic control circuitry utilizes relay logic (coil-to-contact) operation to assure that actuation of a single emergency trip sensor (i.e., one-out-of-two in trip system A or B) will not cause inadvertent closure of the MSIVs. This is consistent with the original design basis of NMP-2 whereby the logic is set up as a one-out-of-two taken twice logic (i.e., one-out-of-two in trip system A and B are required to close the MSIVs). For example, a tripped sensor (reactor low-low water level) provides open contacts to a logic function which causes the sensor relay in the associated trip channel to de-energize. The open contacts from the de-energized sensor relay are connected in logic functions which cause a trip relay to de-energize. Output from the de-energized trip relays are combined in one-out-of-two taken twice logic which generates closure signals for the main steam isolation valves.

Each wye-pattern MSIV contains two electrically operated solenoid valves, a three way pilot solenoid valve with two coils and a test solenoid valve. The two pilot solenoid coils on a MSIV are fed from different trip systems. Since the two (2) trip pilot solenoid coils are supplied power from two (2) different trip systems and both trip systems must de-energize to operate, a transfer and isolation scheme (ball valve) is not required on the wye-pattern valves. This change and the standard General Electric control scheme have reduced the number of field cables.

The new wye-pattern globe valves will use a three position selection switch (close-auto-test) and a pushbutton switch for each valve. These switches are located in the control room and are similar to the ball valve design. The staff concludes that the latest design modifications made to the MSIVs did not change the actuation control system logic or power supplies and is consistent with the original design basis for NMP-2. The MSIV limit switch inputs to the Reactor Trip System logic remain unchanged from the ball valve design. However, the trip setpoint from the ball valve has been changed from 94% open to 92% open for the wye-pattern valve. We reviewed this 2% difference in setpoint and the supporting analysis and find the trip setpoint of 8% acceptable.

Technical Specification Change

The licensee has requested that a revision be made to the NMP-2 Technical Specification Tables 2.2.1-1, 3.6.1.2-1 and 3.6.3-1 to address the installation of the new MSIVs. Table 2.2.1-1 has been changed to account for differences in the physical configuration of the position indicating switches between the ball valves and the new wye-pattern globe valves. Industry experience has indicated that the current Nominal Trip Setpoint of less than or equal to 6% closed cannot be met with the mounting brackets on the globe valves. The licensee has proposed that the MSIV-closure setpoint be less than or equal to 8% closed to allow margin for field adjustment. A corresponding allowable value of less than or equal to 12% closed has also been proposed to account for drift (allowable value). Tables 3.6.1.2-1 and 3.6.3-1 have been changed to alter the valve designations to provide consistent notation for the type of valve installed.

Wye-pattern globe valves are used at the Perry Nuclear Power Plant in the same application and have been approved with setpoints identical to those requested for NMP-2. The staff has found that the change of setpoints is appropriate for wye-pattern globe MSIVs, and that changing of the valve designations is also appropriate.

License Change

Items 1.a.(2), (3), and (4) of Attachment 1 of the License are being deleted. These items are (1) cracked MSIV roller bearings, (2) failure of the MSIVs to close in the required time and (3) failure of the MSIVs to meet local leak rate test requirements. These changes are no longer applicable since they pertain to the ball valves only and these valves have been replaced.

SUMMARY

The proposed change to the MSIV-closure setpoint in Technical Specification Table 2.2.1-1 necessitated by the MSIV change was evaluated against affected transient and accident analyses and the proposed change has been shown not to involve a significant increase in the probability or consequences of an accident previously evaluated. Table 3.6.1.2-1 has been changed to alter the valve designation to provide consistent notation for the type of valve installed, e.g., an air-operated (AOV) valve. The change results from the use of air-operated valves instead of hydraulic-operated valves. Table 3.6.3-1 has also been changed to alter the valve designation to provide consistent notation. For the reasons discussed in this evaluation, we find the proposed changes in Technical Specification Tables 2.2.1-1, 3.6.1.2-1, and 3.6.3-1 are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of the facility components located within the restricted areas as defined in 10 CFR 20. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 15, 1987

PRINCIPAL CONTRIBUTORS:

J. Joyce, ICSB
B. Marcus, ICSB
G. Thomas, RSB
F. Witt, ECEB

TABLE 1

COMPARISON OF LOCA ANALYSIS

	BALL	WYE
Large Break PCT (°F)	1921	1922
Small Break PCT (°F)	1522	1524
Allowable PCT (°F)	2200	2200

COMPARISON OF TRANSIENT ANALYSIS

	BALL	WYE
Operating Limit CPR	1.28	1.28
Safety Limit MCPR	1.06	1.06
Limiting Transient ΔCPR (Load Rejection Without Bypass)(1)(2)	0.22	<0.22
MSIV Closure Event ΔCPR	0.01	<0.01
Peak Vessel Pressure (psi)	1268	1271
Allowable Pressure (psi)	1375	1375

(1) Load rejection without bypass Section 15.2 of the FSAR using ODYN Option A
(2) No change in Limiting Transient

ATTACHMENT 1

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

SAFETY EVALUATION REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO WYE PATTERN GLOBE VALVES (MSIVs) WITHOUT A LEAKAGE CONTROL SYSTEM

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT, UNIT 2

DOCKET NO. 50-410

1.0 INTRODUCTION

The Main Steam Line Isolation Valves (MSIVs) of a Boiling Water Reactor (BWR) are designed to isolate the Reactor Pressure Vessel (RPV) in the event of a design basis steam line break downstream of the MSIVs, a design basis Loss of Coolant Accident (LOCA), or any other event that would warrant containment isolation. This is required by 10 CFR Part 50 Appendix A, General Design Criteria (GDC) 54 and 55. The closure of the MSIVs should terminate releases of radioactivity from the RPV for accidents within the design bases, and ensure that offsite and onsite dose guidelines of 10 CFR Part 100, and 10 CFR Part 50, Appendix A, GDC 19, respectively, are not exceeded.

In April 1979, the licensee selected 24 inch positive seal ball-type MSIVs to replace the wye pattern globe valves and the leakage control system as described in the Nine Mile Point Unit 2 Preliminary Safety Analysis Report. Because of the unique design of the positive valve seal, the leakage from the ball valves was expected to be very low. Therefore, a leakage control system was not considered necessary for the ball-type MSIVs. This conclusion was documented in a letter to G. K. Rhode of Niagara Mohawk Power Corp. from R. L. Tedesco, dated January 2, 1981.

Experience with the ball-type MSIVs during preoperational testing at Nine Mile Point Unit 2 and laboratory prototype testing have failed to demonstrate that these valves will function as anticipated. Delamination of the tungsten carbide coating on the ball was observed, which is believed to have been caused by wearing of the stellite seat. This resulted in excessive seat leakage. Also, during the initial valve test at system temperatures in an offsite prototype facility, packing leakage developed. In view of these engineering problems and a schedular concern, the licensee informed NRC in a letter dated March 11, 1987 (NMP2L 1004), that the Nine Mile Point Unit 2 ball type MSIVs will be replaced with wye pattern globe valves, manufactured by Rockwell, that are similar to those being used in other BWRs. Shop acceptance test results indicate that the wye pattern globe valves leak between 2 and 4 scfh, which meet the Technical Specification limit of 6 scfh (limit for ball type MSIVs). Further, the new valves will close in 3 to 5 seconds, also in accordance with the Technical Specifications.

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By letter dated March 18, 1987 (NMP2L 1007), the licensee requested that a Leakage Control System (LCS) not be required. Regulatory Guide 1.96 "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants", describes a basis for implementing General Design Criteria (GDC) 54 with regard to a leakage control system (LCS) for the MSIVs to ensure that the radiological consequences of design basis accidents do not exceed the dose guidelines of 10 CFR Part 100. The licensee proposed an alternative to a LCS using NUREG-1169 as guidance, pending final resolution of Generic Issue C-8, MSIV Leakage and LCS Failures. A realistic fission product transport model developed by the BWROG in NUREG-1169 was used by the licensee to assess the offsite and onsite dose consequences of alternate means of managing post-accident MSIV leakage using both safety-grade and non-safety-grade systems that could be available for service after a Loss of Coolant Accident (LOCA). The licensee's radiological analysis takes credit for the isolated condenser (main steam line condensate drains open to the condenser) as a MSIV post-accident leakage management method. The analysis demonstrates that the 10 CFR Part 100 dose guidelines will not be exceeded at leakage rates substantially in excess of the Technical Specification limit of 6 scfh. The analysis further indicates that a total MSIV leak rate of 150 SCFH for all main steam lines (38 scfh/steam line) would not result in control room personnel doses in excess of 10 CFR Part 50, Appendix A, GDC 19.

By letter dated March 31, 1987 (NMP2L 1014), the licensee has provided the following additional information requested by the staff:

1. A comparison of the Nine Mile Point 2 Rockwell MSIVs to Rockwell valves used at other nuclear power plants for the intended service;

2. A compilation of industry leak rate testing results and experience;

3. An evaluation of items 1 and 2 above, and a comparison to the NUREG-1169 analysis performed for Unit 2, including an estimate of leakage performance over the first operational cycle; and

4. A discussion of the maintenance practices planned at Unit 2 to enhance low leakage characteristics.

2. EVALUATION

2.1 Leakage Control System

Beginning about 1970, the staff's concern over the possible dose consequences of MSIV leakage at or above the Technical Specification leakage limit led to the requirement that a Leakage Control System be installed in new plants. Until a couple of years ago a majority of the "as found" MSIV leakage values were often in excess of Technical Specification limits. In some cases, MSIV leakage rates were greatly in excess of the Technical Specification value, such that a LCS would have been ineffective because of flow limitations in its design.

As a result of these concerns, the staff prioritized the MSIV leakage and LCS failures as a high priority Generic Issue (C-8). Independently, the BWR Owners Group (BWROG) formed the MSIV Leakage Control Committee to determine the cause of the high leakage rates associated with many of the MSIVs and to develop recommendations to reduce the leakage rates.

The licensee has concluded that a MSIV Leakage Control System is not necessary since Nine Mile Point 2 has a means of collecting, treating, and discharging from the stack MSIV leakage using existing systems. This can be accomplished by:

1. A passive steamline drain system which automatically opens on loss of air power and first stage turbine pressure to the main condenser;

2. Electric boilers capable of providing steam to the steam jet air ejectors, offgas system, and turbine gland seal and exhaust system; and

3. In the event of a LOCA and/or loss of offsite power, NMP2 has the capability to re-establish condenser vacuum, the operation of the steam jet air ejector, the operation of the gland seal and exhaust system, and the offgas steam once offsite power is restored.

2.2 MSIV Leakage Experience

To assess the expected MSIV leakage characteristics for Nine Mile Point Unit 2, leakage tests at operating plants using the Rockwell wye pattern globe valves that are similar to the MSIVs being installed at Nine Mile Point Unit 2 were reviewed. A total of 39% of the "as found" leakage test results were 6 scfh or below, which is the Technical Specification limit for Nine Mile Point Unit 2. Cumulatively, 85% of all test results were less than 38 scfh. In the future the leakage rate percentage below 6 scfh as well as below 38 scfh could conceivably be higher, mainly due to the adoption of the recent BWR Owners Group recommendations. These recommendations include improvements in test methods, maintenance procedures, training and tooling. In addition, all other BWRs have MSIV Technical Specification leak rates of 11.5 scfh or above. The Technical specification limit by itself does not ensure that the refurbished valves will not leak above 6 scfh. The higher Technical Specification leak rates greater than 6 scfh, however, do bias the percentage of leakage rate results below 6 scfh on the low side. It can be concluded that a sound maintenance program should limit valve leakage degradation and increase leakage test results within Technical Specification values.

2.3 MSIV Design Changes

Based on experience with Rockwell designed MSIVs for BWR service, the licensee made MSIV design changes using information provided by other BWR operating plants, valve suppliers, General Electric Co., and an evaluation of Inspection and Enforcement Bulletins, Notices and Circulars applicable to Nine Mile Point Unit 2 MSIVs. These design changes include:

1. Disc-piston connection configuration changed from a spherical backseat to resolve disc-to-piston separation questions;

2. Numatic air valves replaced with Norgren air valves to resolve sticking air valve spools;

3. Improved stem/stem-disc and main disc/piston connection (joints) to resolve stem/stem-disc and main disc to piston separation potential;

4. Spring flange bronze bushing used to reduce the tendancy for galling/ friction between yoke guides and tubes;

5. New spring divider material used to reduce the tendancy for galling/ scoring of the yoke guide tubes; and

6. Modified packing chamber design with graphite rings were used to replace asbestos packing to enhance packing and stem leak lightness capability.

2.4 Radiological Assessment

In the event that leakage values are in excess of the Technical Specification limit, 10 CFR Part 100 offsite and 10 CFR Part 50, Appendix A, GDC 19 control room operator dose guidelines would not necessarily be exceeded. The 39 scfh leak rate, that 85% of the leakage tests met, is important from the standpoint that the calculated doses have been found to be within the controlling design basis accident dose guideline valves of 10 CFR Part 50, Appendix A, GDC 19. Based on NUREG-1169 methodology, and using realistic assumptions of the holdup volume and surfaces of the main condenser and main steamlines and fission product attenuation elsewhere, offsite and control room doses were evaluated. The licensee's analysis indicated that 10 CFR Part 100 offsite doses would be met. The licensee indicated that the control room was limiting and that a combined MSIV leak rate of 150 scfh for all main steam lines (38 scfh per main steam line) would not result in control room personnel doses in excess of 10 CFR Part 50, Appendix A, GDC 19.

On the basis of our review, we conclude that the licensee's radiological evaluation, which takes credit for the isolated condenser, is reasonable. This analysis is a departure from the Standard Review Plan and Regulatory Guides in that some realistic assumptions were utilized for assessing control room habitability (GDC 19) if a design basis LOCA were to occur and the MSIVs leaked at rates in excess of their Technical Specification limit of 6 scfh.

For such an accident during which the MSIVs leaked at rates of 6 scfh, or less, the staff and licensee have both determined that the dose guidelines of GDC 19 would be met. These analyses followed the guidance of the Standard Review Plan with two exceptions. The first exception was the modeling of atmospheric dispersion. The second was credit for post-accident fission product attenuation in the steamlines.

The staff also has reasonable assurance that the 38 scfh leak rate per main steam line represents an upper bound when one considers the expected improved leakage values for the Nine Mile Point Unit 2 valves, provided that effective and careful MSIV maintenance is followed. The licensee expects deterioration in the MSIV leakage to result in leakage rates less than 16 scfh at the end of the first operating cycle.

The overall risks from accident sequences in which MSIV leakage is a significant factor are low without a LCS, and post accident management schemes (including those stated above) were shown to produce significant offsite dose reductions in lieu of a LCS. MSIV leakage was concluded to be a trivial safety concern, and MSIV leakage control was shown to not be risk significant (most of the risk being from accidents resulting in core melt and containment failure). However, for accidents that do not result in containment failure, MSIV leakage can still be important. Several leakage treatment methods which make use of the holdup volume and surface of main steam lines and condensers, and fission product attenuation elsewhere, were evaluated and indicated lower offsite dose consequences than with a LCS. Nine Mile Point Unit 2 design features are similar to the NUREG-1169 base plant and, therefore, the conclusions of NUREG-1169 are considered applicable to Nine Mile Point Unit 2 with a leakage limit of 6 scfh. NUREG-1169 concluded that the low public exposure (isolated condenser and 11.5 scfh leak rate - 5.9×10^{-6} man rem/plant year whole body public exposure; LCS and 11.5 scfh leak rate - 1.0×10^{-4} man rem/plant year whole body public exposure) does not justify a LCS.

2.5 Technical Specification Changes

The replacement wye pattern globe MSIVs are air-operated (AOV) valves and the ball MSIVs were hydraulically-operated. This necessitates valve nomenclature changes in Technical Specification Table 3.6.1.2-1 and 3.6.3-1 from 2MSS*HYV6A, B, C, D and 2MSS*HYV7A, B, C, D to 2MSS*AOV6A, B, C, D and 2MSS*AOV7A, B, C, D.

2.6 MSIV Maintenance and Procedures

The staff is reasonably assured that the wye pattern globe valves being installed in Nine Mile Point Unit 2 without a leakage control system, but with a post accident leakage treatment method, can perform their function without exceeding the dose guideline values of 10 CFR Part 100 and GDC 19 of 10 CFR Part 50. This assurance is dependent on proper maintenance practices and potential operator actions/emergency operating procedures to limit MSIV radioactivity releases. By letter dated April 7, 1987, the licensee has committed to implement the following prior to criticality.

1. vendor recommended maintenance boring, grinding, and lapping tools will be available for refurbishment as needed to restore MSIVs to less than 6 scfh leakage,

2. maintenance procedures based upon MSIV instruction manuals, vendor and General Electric recommendations (including careful maintenance, supervision and inspections to indicate incipient failures);

3. training programs for MSIV maintenance personnel;

4. operating procedures for the post-accident control and treatment of MSIV leakage to limit radioactivity releases as recommended by BWROG in NEDO-30324; and

5. emergency operating procedures to limit radioactivity release through the MSIVs as recommended by the BWROG in NEDO-30324.

3.0 CONCLUSIONS

On the basis of the evaluation above, the staff concludes that Nine Mile Point Unit 2 may commence plant operation without a leakage control system, but with post-accident leakage management. This conclusion is based on a sound MSIV maintenance program committed to by the licensee which includes: maintenance procedures, tooling and equipment, personnel training, operating and emergency procedures, management and inspection.

At the Technical Specification leak rate of 6 scfh, the dose guidelines valves of 10 CFR Part 100 and GDC 19, calculated for a design basis LOCA, will not be exceeded. In the event that the Technical Specification limit is exceeded, 10 CFR Part 100 offsite and 10 CFR Part 50 GDC 10 control room operator dose guidelines calculated using the methodology from NUREG-1169 would not be exceeded.