ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. NPF-69
DOCKET NO. 50-410

Proposed Changes to Technical Specifications

Replace existing page 3/4 6-6 with the attached revised page. This page has been retyped in its entirety with a marginal marking to indicate the change.

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.	24.0
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.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. NPF-69
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Supporting Information and No Significant Hazards Consideration Analysis

INTRODUCTION

During a design basis - loss of coolant accident (DBA-LOCA), radioactive steam is released from a Reactor Recirculation System pipe rupture and pressurizes the drywell. The radioactivity released to the drywell is defined by a source term which specifies the amount and chemical form of the fission product species released from the fuel, as well as the characteristics or timing of the release. The source term is defined by Regulatory Guide (RG) 1.3, Revision 2, entitled "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." Consistent with the guidance provided by this RG, the accident source term is conservatively assumed to be available in the drvwell at the onset of the DBA-LOCA.

Regulatory Position C.1.f of RG 1.3 states; "No credit should be given for retention of iodine in the suppression pool." However, Standard Review Plan (SRP), NUREG-0800, Section 6.5.5, entitled "Pressure Suppression Pool as a Fission Product Cleanup System," replaces Regulatory Position C.1.f of RG 1.3 and provides guidance in the application of fission product scrubbing and retention by the suppression pool. The Staff approved the use of this SRP section at Nine Mile Point Unit 2 (NMP2) for calculating radiological doses in License Amendment No. 56, dated August 30, 1994. As indicated in that license amendment, no credit is taken for suppression pool scrubbing for bypass leakage paths.

Primary containment, which is pressurized due to the LOCA, is assumed to leak radioactivity into secondary containment at the Technical Specification limit, La, of LCO 3.6.1.2.a.1. The Traversing Incore Probe (TIP) System and Engineered Safety Feature (ESF) Systems are also assumed to leak into secondary containment in addition to the primary containment leakage of La. Specifically, the TIP System is assumed to leak radioactivity into secondary containment at the rate of 0.19La. This leakage rate was determined by assuming that one of the five TIP System drive mechanisms was in use and its flexible drive cable became stuck in its associated primary containment penetration at the onset of the LOCA. The penetration was assumed to be unisolated due to a stuck TIP cable in the penetration. No credit was taken for operation of the flexible drive cable's shear valve to isolate the primary containment penetration. The leakage rate was calculated by using conservative friction flow losses through the penetration and by taking credit for the obstruction of flow presented by the stuck cable.

From the onset of the LOCA, primary containment, TIP and ESF leakages are modeled as a ground-level unfiltered release until the establishment of a -0.25 inches water gauge pressure in secondary containment by the Standby Gas Treatment System (SGTS). Thereafter, these leakages, with the exception of bypass leakage paths, are modeled as elevated filtered releases via the SGTS from the main stack.

Bypass leakage is leakage from piping systems that communicate with either the reactor coolant pressure boundary or with the primary containment air space, and terminate outside of secondary containment. Since this leakage bypasses secondary containment, it is unaffected by the filtration of radioactive material in the secondary containment atmosphere afforded by the SGTS. Therefore, bypass leakage is considered an unfiltered ground-level release for the duration of the LOCA. Leakage from the four main steam lines is bypass leakage and is modeled accordingly for radiological purposes.

The bypass leakage paths and their maximum allowable leakage rates are identified in Technical Specification Table 3.6.1.2-1, "Allowable Leak Rates Through Valves in Potential Bypass Leakage Paths." The radiological consequences of the total bypass leakage of this table are added to the radiological consequences of primary containment leakage La, to determine the overall radiological consequences of a DBA-LOCA. Each of the four main steam lines, which are identified in this table, are currently assumed to leak at 6.0 scfh.

Primary containment and the TIP System are assumed to leak into secondary containment at their above leak rates for the duration of the accident, including the drawdown period. Similarly, ESF leakage of suppression pool water to secondary containment is assumed to occur at a fixed rate for the duration of the accident.

Bypass leakage occurs through containment isolation valves in piping routed from primary containment to the outside of the secondary containment. The leakage is caused by the drywell pressure; therefore, long-term drywell temperature and pressure response are considered in calculating the bypass leakage rates. Specifically, bypass leakage paths are modeled as isentropic flow as described in Section 6.2.3.2.4 of the Updated Safety Analysis Report (USAR). This approach varies from that of containment and TIP leakage, but has been reviewed, and accepted, by the Staff, as documented in NUREG-1047, entitled "Safety Evaluation Report related to the operation of Nine Mile Point Nuclear Station, Unit No. 2," (SER), Section 6.2.3.1, Supplement 2.

Elemental and particulate iodines are assumed to plate-out on the internal pipe surfaces prior to leaking from the pipes. The fraction of iodine exiting the pipes is the plateout factor. The radiological analysis of a DBA-LOCA applies iodine plateout factors to the thyroid doses. However, iodine plateout factors are not applied to the gamma and beta doses.

Section 15.6.5 of the USAR presents the radiological evaluation of a DBA-LOCA at NMP2. The doses depicted in the USAR have not been updated to include the recent changes to the radiological analysis approved by the Staff in License Amendment No. 56. In addition, the USAR doses have not been updated to reflect the effects of the proposed power uprate of NMP2 currently under review by the Staff (see NMPC letter to the NRC dated July 1,1994, NMP2L 1397). Otherwise, the assumptions and methodology used to calculate the radiological doses from a DBA-LOCA at NMP2 are presented in USAR Section 15.6.5.

Specifically, License Amendment No. 56 revised the Technical Specifications' secondary containment drawdown time and inleakage testing requirements (Surveillance Requirements (SR) 4.6.5.c.1 and 4.6.5.c.2). The changes to these SRs supported a revised design basis radiological analysis which supported an increase in the drawdown time by taking credit for fission product scrubbing and retention in the suppression pool which was not assumed in the original radiological analysis. Furthermore, the revised analysis took credit for mixing of primary containment and ESF leakages with 50% of the secondary containment free air volume prior to the release of radioactivity to the environment. The revised analysis also reflected the proposed power uprate of NMP2 described above.

The current radiological doses from a DBA-LOCA, which include License Amendment No. 56 and the proposed power uprate, are shown below in Table 1.

TABLE 1

SUMMATION OF LOCA DOSES(1) FOR NMP2

			Whole Body (rem)	Thyroid (rem)	Skin (rem)
EAB (2 hours)	A)	10 CFR 100 Limit	25.0	300.0	* (2)
	B)	Current Dose(4)	4.5	22.8	3.2
	C)	New Dose(5)	4.3	22.4	3.2
LPZ (30 days)	A)	10 CFR 100 Limit	25.0	300.0	* (2)
	B)	Current Dose(4)	2.4	35.9	1.8
	C)	New Dose(5)	2.6	58.3	2.2
Control Room (30 days)	A)	GDC 19 Limit	5.0	30.0	75(3)
	B)	Current Dose(4)	1.6	14.0	22.8
	C)	New Dose(5)	2.0	24.8	34.0

(1) LOCA doses reflect both power uprate and License Amendment No. 56.
(2) No limit specified.
(3) Credit is taken for the use of protective clothing and eye protection during severe radiation releases consistent with the guidance of Section 6.4 of NUREG-0800.
(4) Includes maximum allowable leakage of 6.0 scfh for MSIVs.
(5) Includes maximum allowable leakage of 24.0 scfh for MSIVs.

These current radiological doses from a DBA-LOCA for the currently assumed MSIV leakage of 6.0 scfh are below the guideline values of 10 CFR 100 and General Design Criterion (GDC) 19 of 10 CFR 50, Appendix A.

The control room is normally maintained at a positive pressure with respect to surrounding air volumes by its normal ventilation system using outside air. Upon detection of radioactivity in the air intake path of this normal ventilation system, the air intake path is diverted automatically to the intake of the two Outdoor Air Special Filter Trains, and pressurization is continued by drawing filtered air into the control room. Two separate outside air intakes are available and redundant radiation monitors are located in ductwork common to both intakes.

Operators, in the event of an accident, have the capability of selecting the one intake which provides less contaminated air. Selection of an air intake is done by a control room operator manually positioning an air operated damper from within the control room pressure envelope. The operator can make the proper selection of an air intake by monitoring the response of the radiation detectors as each intake is individually selected and by knowledge of the wind direction and accident-release point. The Staff acknowledged this capability in NUREG-1047, SER section 6.4. Notwithstanding the above, the radiological analysis assumes the more contaminated air intake provides air to the Outdoor Air Special Filter Trains.

This application for amendment to the NMP2 operating license proposes changes to Technical Specification Table 3.6.1.2-1. Specifically, the maximum allowable leak rate through each of the four main steam lines is being increased to 24.0 scfh. The above radiological analysis has been revised to reflect these proposed changes and is discussed below.

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

NMPC proposes a change to Table 3.6.1.2-1, entitled "Allowable Leak Rates Through Valves in Potential Bypass Leakage Paths." The current version and the proposed changes are:

Current version of Table 3.6.1.2-1, page 3/4 6-6

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
.....			

* Test conditions: ai: medium, 40 psig.

Proposed changes to Table 3.6.1.2-1, page 3/4 6-6

LINE DESCRIPTION	VALVE MARK NO	TERMINATION REGION	PER VALVE* LEAK RATE, SCFH
4 Main Steam Lines	?MSS*AOV6A, B, C, D 2MSS*AOV7A, B, C, D	Turbine Bldg.	24.0
.....			

* Test conditions: air medium, 40 psig.

EVALUATION

The current radiological doses from a DBA-LOCA have been revised using the radiological dose methodology described in USAR Section 15.6.5. The new doses, as shown on Table 1 of this attachment, reflect the following:

1. An increased allowable MSIV leakage of 24.0 scfh,

2. License Amendment No. 56 and

3. The proposed power uprate.

The revised radiological doses have increased but continue to remain below 10 CFR 100 guideline values and GDC 19 criteria. The affect of the increased allowable MSIV leakage on the recalculated doses is partially offset by the following changes to the radiological analysis:

1. Credit is taken for fission product scrubbing and retention in the suppression pool for bypass leakage through the four main steam lines and the two feedwater lines. This application of scrubbing to these leakage paths is identical to the application of scrubbing to primary containment leakage during the drawdown period as presented to the Staff in NMPC's letter dated July 1,1994 (NMP2L 1480) and approved by the Staff in License Amendment No. 56, dated August 30, 1994.

2. TIP System leakage has been recalculated to be 0.11La. This recalculated value reflects the use of a more realistic yet conservative friction loss for primary containment flow through an unisolated TIP System primary containment penetration.

3. The Staff recognized the capability to select the less contaminated intake to reduce doses in Section 6.4 of NUREG-1047, SSER2. Consistent with this capability, credit is taken for the selection of the less contaminated air intake path by the control room operators for operation of the Outdoor Air Special Filter Trains. Conservatively, this credit is applied starting eight hours after the onset of the DBA-LOCA. The revised analysis, by necessity, uniformly applies X/Q values for the operating air intake to all release paths, i.e., it does not selectively evaluate each release point such that a combination of east and west air intakes would be credited. Plant procedures will be revised as appropriate to ensure that the less contaminated air intake path is selected by control room operators.

   Selection of the less contaminated air intake in the radiological analysis is based on the following evaluation:

   a. Each air intake has two safety-related dampers in the duct work upstream of the point where the east and west intakes combine. The east air intake is equipped with dampers 2HVC*AOD 61A and 2HVC*DMPV 83. Likewise, the west air intake is equipped with dampers 2HVC*AOD 61B and 2HVC*DMPV 84. One of the two dampers in each duct is air-operated, is normally open, and fails open on loss of air. None of the air operated dampers receives any automatic signal following a LOCA. The second damper in each duct is a balancing damper, is normally open and is locked in place.

b. All of the above dampers are located within the control room ventilation envelope and are easily accessible. Therefore, both physical and radiological post-LOCA access is assured.

c. The capability to manually open or close any of the four dampers has been confirmed to be

- a short period of time (less than 30 minutes) and is
- an easily implementable evolution.

d. Isolation of one of the control room air intakes to support the design basis radiological analysis would not be required until eight hours post-LOCA. This provides ample time for evaluation of release point(s), wind direction and manipulation of the dampers.

e. The probability of a spurious closure of the air operated damper in the less contaminated intake within 30 days following a DBA-LOCA has been evaluated in a plant specific probablistic risk assessment and determined to be approximately 1.2×10^{-14}. Similarly, for an accident involving any core damage, the probability is 4.5×10^{-10}.

In addition, the probability of the failure of the control room operators to close the correct air operated damper within 30 days following a DBA-LOCA has been evaluated in a plant specific probablistic risk assessment and determined to be approximately 9.5×10^{-13}. Similarly, for an accident involving any core damage, the probability is 3.6×10^{-8}.

These probabilities are sufficiently low such that the failure to have the less contaminated air intake available post-LOCA is not considered credible.

f. NMPC, after evaluating the balancing dampers (2HVC*DMPV 83 and 84), has concluded that the balancing dampers are sufficiently leak tight to effectively divert air flow to the desired intake. In fact, an approximately 60% and 40% split in air supply between the less contaminated and the more contaminated intake, respectively, is needed to meet the regulatory dose limits of 10 CFR 100 and GDC 19. The design of the balancing dampers is expected to provide sufficient leak tightness that, based on pressure differentials and frictional losses, essentially all air flow will be diverted to the desired air intake.

Based on items (a) through (f) above, NMPC has concluded that there is no credible single active failure that would prevent selection of the single desired intake.

4. Iodine plateout factors have been previously applied to thyroid doses resulting from bypass leakage. The revised analysis applies iodine plateout credits to beta and gamma doses resulting from MSIV and feedwater bypass leakage.

The low population zone (LPZ) and control room doses in Table 1 are calculated assuming that the primary containment leaks into secondary containment at its Technical Specification limit for 30 days. Actual leakage from this path can approach this value for 30 days only if the primary containment accident pressure remains at its peak value of approximately 40 psig for the 30 day period. This leakage assumption is very conservative in that the operators, as directed by the emergency operating procedures, will be cooling and thereby depressurizing primary containment shortly after water level has stabilized above the top of the active fuel in the reactor pressure vessel. These operator actions will significantly reduce leak rates, and therefore radiological consequences significantly below the design basis calculated doses. In addition, the exclusion area boundary (EAB), LPZ and control room calculated radiological doses all continue to reflect the conservative fuel failure and instantaneous release assumptions of RG 1.3.

Bypass leakage paths continue to be modeled as isentropic flow as described in Section 6.2.3.2.4 of the USAR. As previously indicated, the Staff reviewed and accepted this modeling approach in Section 6.2.3.1 of NUREG-1047, SSER2.

The EAB doses shown on Table 1 are essentially unaffected by the increase in allowable MSIV leak rate because, even at the higher leak rate, the leakage is not postulated to reach the environment until well after 2 hours. EAB doses are calculated for the 0 to 2 hour period of the LOCA.

The LPZ and Control Room whole body doses primarily reflect the increased MSIV leakage rate. Thyroid doses also reflect this increase but are somewhat mitigated by credit applied for suppression pool scrubbing of elemental and particulate iodines from the MSIV and feedwater leakage paths. Beta skin doses are further mitigated by application of iodine plateout credits that have not been previously applied. Control Room doses are further mitigated by selection of the less contaminated outside air intake for periods greater than 8 hours post-LOCA. Control Room whole body dose due to direct shine contribution has not previously included iodine plateout and suppression pool scrubbing credits. Consistent with this approach, these credits are still not applied to direct shine doses.

The impact of the increase in MSIV leakage on vital area access, and equipment qualification has been evaluated. The MSIV leakage path has very low contribution to the NMP2 vital area doses. In addition, other changes made to the DBA-LOCA radiological analysis, as described above, have reduced the vital area doses. Based on these factors, vital area access is maintained, and, in fact, only minimally impacted. Similarly, neither MSIV leakage nor other design basis assumption changes, as described above, affect equipment qualification analyses, and therefore equipment qualification is not impacted.

Therefore, based on the above evaluation, NMPC has concluded that the increase in MSIV leakage from 6.0 scfh to 24.0 scfh will result in radiological doses from a DBA-LOCA which remain below the guideline values of 10 CFR 100 and GDC 19 limits.

CONCLUSION

A Technical Specification change is proposed to Table 3.6.1.2-1, entitled "Allowable Leak Rates Through Valves in Potential Bypass Leakage Paths." Specifically, the change to this table would allow a maximum leakage of 24.0 scfh for each of the eight MSIVs. The current Technical Specifications allow a maximum leakage for an MSIV of 6.0 scfh.

The revised radiological analysis incorporates a maximum leakage of 24.0 scfh for each of the four main steam lines. In addition, the radiological analysis includes the proposed power uprate of NMP2 and the recently issued License Amendment No. 56. The radiological doses from a DBA-LOCA remain within the guidelines of 10 CFR 100 and GDC 19 criteria. The impact of the increased MSIV leakage on vital area access and equipment qualification is minimal and acceptable.

For these reasons, there is reasonable assurance that operation of NMP2 in the proposed manner will not endanger the public health and safety , and that issuance of the proposed amendment will not be inimical to the common defense and security.

NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis using the standards in 10 CFR 50.92 concerning the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91, the following analysis has been performed:

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to Technical Specification Table 3.6.1.2-1 would allow a maximum leakage of 24.0 scfh for each of the eight MSIVs. The current Technical Specifications allow a maximum leakage for an MSIV of 6.0 scfh.

Closure of one or more of the MSIVs at rated power is a pressure transient for the reactor coolant pressure boundary. This pressure transient is evaluated in Section 15.2.4 of the USAR. Closure of MSIV(s), as analyzed in the USAR, could occur due to manual or automatic actions. A change to the leakage limit for the MSIVs does not affect either the manual or automatic actions that would close the MSIVs. Therefore, the proposed change to the table cannot affect the probability of the closure of one or more MSIVs at rated power.

The radiological evaluation of the DBA-LOCA incorporates a maximum leakage of 24.0 scfh for each of the four main steam lines. In addition, the revised radiological evaluation includes the impact of the proposed license amendment currently under review by the Staff which would increase the rated operation of NMP2 from 3323 to 3467 megawatts thermal (see NMPC letter dated July 22, 1993 to the NRC). The revised radiological evaluation also includes the impact of License Amendment No. 56 (see NMPC letter dated July 1, 1994 to the NRC and License Amendment No. 56, dated August 30, 1994).

The new doses from the revised radiological analysis for a DBA-LOCA, as shown in Table 1, continue to remain below 10 CFR 100 guideline values and GDC 19 limits. The impact of the increased MSIV leakage on vital area access and equipment qualification is minimal and acceptable. Therefore, operation with the proposed change to the Technical Specifications will not significantly increase the consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The safety function of the MSIVs is to isolate the main steam lines in a timely manner to preclude the uncontrolled leakage of radioactive steam. This is accomplished by providing the MSIVs with the capability of rapidly closing automatically in response to various plant conditions. The increase in the leakage limit for the MSIVs from 6.0 scfh to 24.0 scfh will not inhibit the MSIVs' isolation function. Therefore, operation with the proposed increase in MSIV leakage will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The revised radiological analysis follows the very conservative fuel failure and instantaneous release assumptions of RG 1.3, with the exception of regulatory position C.1.f as permitted by SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup." The Staff approved the use of SRP Section 6.5.5 as part of the licensing basis of NMP2 in License Amendment No. 56.

The revised radiological analysis incorporates the maximum allowable leakage limit of 24.0 scfh for each of the four main steam lines. The revised radiological analysis also includes the impacts of the proposed power uprate of NMP2 and License Amendment No. 56. The new doses from the revised radiological analysis remain below the Staff acceptance criteria of 10 CFR 100 guideline values and GDC 19 (see Table 1). Therefore, operation with the proposed changes to the Technical Specifications will not significantly reduce a margin of safety.

Accordingly, as determined by the analysis above, this proposed amendment involves no significant hazards consideration.