ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. NPF-69

DOCKET NO. 50-410

Proposed Changes to Technical Specifications

Replace existing pages 2-3 and 3/4 3-17 with the attached revised pages. In addition, insert new page 3/4 3-19a. These pages have marginal markings to indicate the proposed changes.

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TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Intermediate Range Monitor, - Neutron 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[(b)] - 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.

(b) See footnote (**) to Table 3.3.2-2 for trip setpoint during hydrogen addition test.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Primary Containment Isolation Signals (Continued)		
a. Reactor Vessel Water Level*		
1) Low, Low, Low, Level 1	≥17.8 in.	≥10.8 in.
2) Low, Low, Level 2	≥108.8 in.	≥101.8 in.
3) Low, Level 3	≥159.3 in.	≥157.8 in.
b. Drywell Pressure - High	≤1.68 psig	≤1.88 psig
c. Main Steam Line		
1) Radiation - High **	≤3x Full Power Background	≤3.6x Full Power Background
2) Pressure - Low	≥766 psig	≥746 psig
3) Flow - High	≤103 psid	≤109.5 psid
d. Main Steam Line Tunnel		
1) Temperature - High	≤165.7°F	≤169.9°F
2) ΔTemperature - High	≤ 66.7°F	≤ 71.3°F
3) Temperature - High MSL Lead Enclosure	≤146.7°F	≤150.9°F
e. Condenser Vacuum Low	≥8.5 in Hg vacuum	≥7.6 in. Hg vacuum
f. RHR Equipment Area Temperature - High (HXs/A&B Pump Rooms)	≤135°F	≤144.5°F
g. Reactor Vessel Pressure - High (RHR Cut-in Permissive)	≤128 psig	≤148 psig
h. SGIS Exhaust - High Radiation	≤5.7×10^{-3} μCi/cc	≤1.0×10^{-2} μCi/cc

Table 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

** Within 24 hours prior to the planned start of the hydrogen injection test and with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours after completion of the hydrogen injection test. At reactor power levels below 20% rated power hydrogen injection shall be terminated, and control rod withdrawal is prohibited until the Main Steam Line Radiation Monitor trip setpoint is restored to its pre-test value.

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Supporting Information and No Significant Hazards

Consideration Analysis

Discussion

The primary recirculation coolant has a neutral pH and contains 100 to 300 ppb dissolved oxygen with stoichiometric amounts of hydrogen. The concentration of oxygen increases the susceptibility of austenitic stainless steel to Intergranular Stress Corrosion Cracking (IGSCC) when other requisite factors such as stress and sensitization are present. Reduction or elimination of the oxygen in the coolant will reduce or eliminate the potential for IGSCC. The proposed approach to reduce oxygen concentrations at Nine Mile Point Unit 2 (NMP2) is to adopt the Hydrogen Water Chemistry (HWC) conditions where added hydrogen will be used to suppress radiolytic oxygen formation. A pre-implementation test is scheduled for the second quarter of 1991 to demonstrate the feasibility of HWC for NMP2 and to monitor plant parameters during hydrogen addition.

The addition of hydrogen reduces the concentration of oxygen in the reactor coolant and increases the carryover of N^{16} in the steam. The increased presence of N^{16} is expected to increase the background radiation level seen by the Main Steam Line Radiation Monitor (MSLRM) by a factor of approximately five for peak hydrogen concentration. The increased radiation level will necessitate a temporary change in the steam line radiation monitoring setpoint required by Tables 2.2.1-1 and 3.3.2-2 of the Technical Specifications in order to avoid spurious reactor trips.

The function of the MSLRM is to detect gross fuel failure as indicated by increased amounts of fission products in the steam piping and to initiate a Reactor Protection System (RPS) trip and MSIV isolation. The MSLRM trip setting is set high enough above the background radiation level to prevent spurious reactor trips yet low enough to detect gross fission product release. USAR transient analyses do not take credit for a MSLRM initiated trip.

Therefore, increasing the MSLRM setpoint will not affect any of the Chapter 15 transient analyses. In the Updated Safety Analysis Report (USAR) accident analyses, only the radiological consequences of the Control Rod Drop Accident (CRDA), addressed in USAR Section 15.4.9, can be affected by the availability of the MSLRM to provide a signal to initiate MSIV closure. The analyses assumes that the MSIVs receive an automatic closure signal 0.5 seconds following the detection of high radiation in the main steam lines. The USAR CRDA analyses do not take credit for reactor trip from a MSLRM signal in assuring that the design limit for peak fuel enthalpy of 280 cal/gm is not exceeded.

The radiological consequences from a postulated CRDA are more severe for power levels below 10% of rated power. Above this power level the rod worth and resultant CRDA peak fuel enthalpies are not limiting due to core voids and faster Doppler feedback (see Reference 1). The licensing basis for the CRDA as described in USAR Section 15.4.9.3.1 states that the maximum control rod worth is established by assuming the worst single inadvertent operator error. From References 2 and 3, the maximum control rod worth above 20% rated power, assuming a single operator error, is <0.5% ΔK/K. Parametric studies utilizing the conservative GE excursion model (Reference 1) indicate that the maximum peak fuel enthalpy for a dropped control rod worth of 0.8% ΔK/K is less than 120 cal/gram (Reference 3). Consequently, the conservatively calculated peak fuel enthalpy for a CRDA above 20% rated power will have significant margin to the fuel cladding failure threshold of 170 cal/gram.

In order to maintain the margins discussed above, the MSLRM setpoint changes will occur only at reactor power levels above 20% rated. In the event that reactor power unexpectedly decreases below 20%, the proposed changes prohibit control rod withdrawal without first restoring the MSLRM setpoint to its pre-test value. Therefore, there will be no impact on the design basis radiological effects listed in USAR Table 15.4-13. In summary, increasing the Full Power Background in the MSLRM setpoint will not alter the conclusions reached in Chapter 15 transient and/or accident analyses.

The MSLRM setpoint adjustment itself will not significantly increase the probability of an inadvertent scram or transient. Half scram and half MSIV (Group 1) isolation signals may be temporarily introduced as individual MSLRM setpoints are adjusted for hydrogen water chemistry operating conditions in accordance with Technical Specifications 3.3.1 and 3.3.2. The time the plant could be operating in this condition is not significant. Furthermore, the hydrogen test requires increased monitoring and evaluation of the MSLRM instrument and MSLRM trip setpoint.

Plant capability to monitor and detect potential fuel defects and failure will be maintained by (a) the Off-gas Radiation Monitors, (b) the Main Steam Line Radiation Monitoring scram and isolation system, and (c) the performance of daily primary coolant isotopic water analyses.

In order to meet ALARA requirements, the following additional protective measures will be taken during the hydrogen injection periods:

1. Hydrogen addition will occur during shift periods that will minimize personnel radiation exposures.

2. Locations where increased radiation levels during the test period are expected will be identified. Restricted access and/or additional shielding will be implemented as appropriate for these locations.

3. Radiation level surveys will be taken at various hydrogen flow rates.

4. Area radiation monitors will be logged at specific increments of hydrogen addition.

5. If at any time the radiation increase is significantly higher than projected or unanticipated problems with interfacing systems are encountered, the hydrogen addition rates will be reduced to the previous step. If this does not resolve the concern, the test will be terminated. Due to the short half-life of N^{16} (7.1 seconds), radiation levels will return to the pre-test conditions within minutes of hydrogen shutoff. The test will not be resumed until a thorough evaluation of the situation is completed.

6. Site surveys will be conducted to measure N^{16} gamma.

Areas where radiation levels will temporarily increase above 1000 mrem/hr during the hydrogen addition will be posted, roped off, and provided with a flashing light in compliance with the requirement in Section 6.12.2 for high radiation areas where no enclosure exists.

The additional radiation protection measures described above are in compliance with the requirements of 10CFR20.1(c) and Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposure At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." These measures will be implemented during the test in addition to established applicable

NMP2 procedures. Therefore, there will be no reduction in the effectiveness or level of the radiation protection program at NMP2.

10CFR50.91 requires that at the time a licensee requests an amendment, it must provide the Commission its analysis using the standards in 10CFR50.92 concerning the issue of no significant hazards consideration. Therefore, in accordance with 10CFR50.91, 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 do not involve a significant increase in the probability or consequences of an accident previously evaluated. The Control Rod Drop Accident (CRDA) in the Updated Safety Analysis Report (USAR) Section 15.4.9 is the only accident that references the MSIV closure due to high radiation. Per Reference (1) the CRDA is not a concern above 10% rated power. Above 10% rated power, the combined effects of increased core voiding and faster Doppler feedback assures that peak fuel enthalpy and rod worths will remain well below design limits for a CRDA. The maximum rod worth for a CRDA is a result of a single operator withdrawal error. At power levels above 20%, a single operator control rod withdrawal error will not produce sufficient rod worth to exceed design peak fuel enthalpy. Accordingly, in the proposed change the Main Steam Line Radiation Monitor High Radiation Setpoint will not be increased unless rated power is above 20% of rated. In addition, below 20% rated power, control rod withdrawal is prohibited and hydrogen injection is terminated. The setpoint must be returned to its pretest value for control rod withdrawal below 20% rated power. The power requirements of the proposed change are consistent with the above argument and assures that the probability or consequences of a CRDA are not significantly increased.

During the brief time the MSLRM trip and alarm setpoints are adjusted, a half scram and half MSIV (Group 1) isolation may be temporarily introduced as each individual MSLRM drawer is set to the hydrogen water chemistry operating condition in accordance with Technical Specifications 3.3.1 and 3.3.2. The time the plant could be operating in this condition due to the setpoint adjustment is negligible. In addition, the hydrogen addition test requires increased observation and evaluation of the MSLRM setpoint at incremental test periods such that the MSLRM trip setpoint will not be exceeded.

Therefore, the setpoint adjustment will not significantly increase the probability of an inadvertent scram or transient.

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 previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not initiate any new accidents not previously considered in the USAR or prevent any component from performing its design basis safety function. The proposed increase in the full power background radiation levels during the hydrogen addition test does not prevent the MSLRM from performing its intended function of detecting a gross fission product release in the event of a CRDA.

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.

Below 20% rated power the MSLRM setpoint shall be returned to its pre-test value. Consequently, the Technical Specification Bases and required design function of the MSLRM will remain valid. Above 20% of rated power inherent reactivity mechanisms limit the consequences of a CRDA such that a substantial margin to the fuel cladding failure threshold will be maintained. Therefore, the proposed change will not involve a significant reduction in the margin of safety.

References

1. R. C. Stirn et al. "Rod Drop Analysis for Large Boiling Water Reactors." NEDO-10527. General Electric Company, March 1972.

2. R. C. Stirn et al. "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum no. 2 Exposed Cores." NEDO-10527, Supplement 2. General Electric Company, January 1973.

3. R. C. Stirn et al. "Rod Drop Analysis for Large Boiling Water Reactors Addendum No.1 Multiple Enrichment Cores with Axial Gadolinium." NEDO-10527 Supplement 1. General Electric Company, July 1972.