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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 Introduction

By letter[1] dated November 21, 1978 and supplemented by letters[2,3] dated January 2, 1979 and February 12, 1979, the Niagara Mohawk Power Corporation (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-63 for Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1). The proposed changes relate to the seventh refueling of NMP-1, involving the replacement of 76 exposed 7x7 fuel assemblies and 108 exposed 8x8 fuel assemblies with an equivalent number of fresh, two water rod, retrofit 8x8 fuel assemblies designed and fabricated by the General Electric Company. In support of this reload application for NMP-1, the licensee has submitted supplemental reload licensing documents[4,5] prepared by the General Electric Company (GE), proposed plant Technical Specification changes[6] and provided responses[3] to our request[7] for additional information on the reload application.

This reload (Reload 7) is the first for NMP-1 to utilize GE's retrofit 8x8R fuel design, although several other operating BWRs have already refueled with the new GE fuel design. Additionally, four lead retrofit 8x8 test assemblies, previously loaded into an operating reactor core, have performed satisfactorily for at least two cycles.

The descriptions of the nuclear and mechanical design of the replacement 8x8R fuel assemblies and the exposed standard 8x8 fuel assemblies, which were used in connection with the most recent NMP-1 reloads, are contained in GE's generic licensing topical report[8] for BWR reloads. Reference 8 contains a complete set of references to other GE topical reports which describe GE's BWR reload methods for the nuclear, mechanical, thermal-hydraulic, transient and accident analysis calculations. Information addressing the applicability of these methods to reload cores containing both 8x8 and 8x8R fuel is also contained in Reference 8. Portions of the plant-specific data, such as operating conditions and design parameters, used in transient and accident calculations, have also been included in the topical report.

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Our safety evaluation[9] of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was provided in the staff's evaluation[10] of the information contained in Reference 11.

As part of our evaluation[9] of Reference 8 we found the cycle-independent input data for the reload transient and accident analyses for NMP-1 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 4, which follows the format and content of Appendix A of Reference 8.

As a result of our generic evaluation [9] of a substantial number of safety considerations relating to the use of 8x8R fuel in mixed core loadings with 8x8 fuel, only a limited number of additional review items are included in this evaluation of Cycle 6 of NMP-1. These include the plant and cycle-specific input data and results presented in References 4 and 5, the LOCA-ECCS analysis results for the reload fuel design, and those items identified in our evaluation[9] as requiring special consideration during reload reviews.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 6, up to 184 fresh 8x8R fuel bundles, with a bundle average enrichment of 2.77 wt/% U-235 will be loaded into the core, replacing a like number of exposed 7x7 and 8x8 assemblies. The remainder of the 532 fuel assembly reconstituted core will consist of irradiated 8x8 fuel assemblies exposed during Cycles 4 and 5. Thus, about 35 percent of the fuel bundles are being replaced for this reload. The reference core loading for Cycle 6 will result in eighth core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of References 4 and 5 indicates that the fuel temperature and void dependent characteristics of the refueled core are not significantly different from previous cycles of NMP-1. Additionally, scram effectiveness, as shown in Figures 2a, 2b and 2c of References 4 and 5, is also similar to earlier cycles. The 1.2%Δk/k calculated shutdown margin for the reconstituted core meets

the Technical Specification requirement that the core be subcritical by at least 0.25%Δk/k in the most reactive operating state when the single most reactive control rod is fully withdrawn and all other rods are fully inserted. Finally, Reference 4 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 3.6%Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 9, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. When meeting this MCPR safety limit, during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 6 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 5. The basis for the revised safety limit is addressed in Reference 8, while our generic approval of the new limit is given in Reference 9.

2.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been conservatively analyzed for both the exposed 8x8 fuel and the reload 8x8R fuel at the most adverse cycle exposure condition.

The methods used for these calculations, including cycle-independent initial conditions and transient input parameters are described in Reference 8. Our acceptance of the values used and related transient analysis methods appears in Reference 9. Supplementary cycle-dependent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of References 4 and 5. Our evaluation of the methods used to develop these supplementary

transient input values have already been addressed and appear in Reference 8. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis[9] for the determination of transient ΔCPRs. The transient events analyzed were: pressurization (turbine trip without bypass, and feedwater controller failure), feedwater temperature reduction (loss of 100°F feedwater heating) and local reactivity insertion (control rod withdrawal error).

The licensee reports that the most limiting event in the above categories for both the exposed 8x8 assemblies and the reload 8x8R assemblies is the control rod withdrawal error. This transient results in CPR reductions of 0.28 for the standard 8x8 assemblies and 0.30 for the retrofit 8x8 assemblies, with an Average Power Range Monitor rod block setpoint of 105%. Addition of these ΔCPRs to the 1.07 SLMCPR establishes fuel type dependent operating limit MCPRs (i.e. 1.35 for 8x8 fuel and 1.37 for 8x8R fuel) sufficient to assure that the SLMCPR will not be violated during Cycle 6 even if any of the aforementioned events were to occur.

The licensee also has considered the effects of possible fuel loading errors (FLE) on bundle CPR. The results of the licensee's FLE analysis (see Section 2.3.3 herein) shows that a somewhat higher MCPR operating limit would be required for the 8x8 assemblies in order to assure that the MCPR safety limit would not be violated in the event of the most severe FLE. In view of these results, the licensee has proposed that for Cycle 6, the 8x8 MCPR operating limit be adjusted upward from the aforementioned 1.35 to 1.40. These operating limits MCPRs (i.e., 1.40 for the 8x8 bundles and 1.37 for the 8x8R bundles) are acceptable to the staff.

2.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawal error and fuel loading error events were analyzed by the licensee, to determine the maximum linear heat generation rates (LHGR). The results for NMP-1 Cycle 6 show that the fuel type and exposure-dependent safety limit LHGRs, shown in Table 2-3 of Reference 6 will not be violated should these events occur. Thus, fuel failure due to excessive cladding strain will be precluded should either of these events occur. These results are acceptable to the staff.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for

Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading... "the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

For Cycle 6 the licensee has re-evaluated the adequacy of NMP-1 ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these plant-specific analyses are given in Reference 4.

We have reviewed the information submitted by the licensee and conclude that NMP-1 will be in conformance with all the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when operated in accordance with the MAPLHGR versus Average Planar Exposure values appearing in Section 14 of Reference 4.

2.3.2 <u>Control Rod Drop Accident</u>

The key plant-specific nuclear characteristics for the worst case control rod drop accident (CRDA) occurring during either cold startup or hot startup conditions are within the values used in the bounding CRDA analysis given in Reference 8. The bounding analysis shows that the peak fuel enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 6 of NMP-1 the peak fuel enthalpy associated with a CRDA from either cold or hot startup conditions will also be within the 280 cal/gm design limit.

2.3.3 <u>Fuel Loading Error</u>

The licensee has considered the effect of postulated fuel loading errors on bundle CPR. An analysis of the most severe fuel loading errors were performed using GE's standard methods, which have previously been reviewed and approved by the staff. The results show that worst possible fuel bundle misloadings will not cause a violation of the 1.07 safety limit MCRP assuming the proposed 1.40 OLMCPR for the 8x8 fuel assemblies and 1.37 OLMCPR for the 8x8R fuel assemblies. Thus,

these operating limits MCPRs will effectively preclude DNB related fuel failures caused by either fuel cladding overheating or cladding oxidation, which might otherwise occur because of a fuel loading error accident. This is acceptable to the staff.

2.4 Overpressurization Analysis

For Cycle 6 the licensee presented[4] the results of an overpressurization analysis in order to demonstrate that adequate margin exists to the ASME code allowable vessel pressure (110 percent of vessel design pressure). The transient analyzed was the closure of all main steam isolation valves with no reactor scram. The analysis was performed assuming 100 percent power, core nuclear physics parameters applicable to the end of Cycle 6, no credit for the relief function of the safety/relief valves, no reactor scram and all safety valves operative. The results of this analysis, postulated to occur during the most adverse time during the cycle, shows that the peak pressure at the vessel bottom would be 1315 psig. This provides a 60 psi margin to the 1375 psig ASME code limit.

Overpressure analyses accepted by the staff on other BWR reload applications have assumed MSIV closure with high neutron flux scram and one failed safety valve. However, the assumption of no scram for the overpressurization analysis for Reload 7 of NMP-1 represents a conservatism which we believe more than compensates for the assumption of no failed safety valve. Thus, the staff finds the 60 psi pressure margin to the 1375 psig ASME code allowable limit to be acceptable.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 8. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state (corresponding to the intersection of the natural circulation curve and the extrapolated rod block line) are 0.46 and 0.51 respectively. These are both well below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from higher power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay

ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. GE is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

Although we have not yet arrived at a final generic evaluation of GE's BWR stability methods and design criteria, in view of the relatively low decay ratios calculated for this reload together with the methods qualification information submitted by GE to date, we find the stability margins for Cycle 6 of NMP-1 to be acceptable.

2.6 Pressure Margin to Safety Valve Actuation

GE currently recommends[8] that for the most severe abnormal operational transient, a 25 psi margin be maintained to the lowest safety valve setpoint. The purposes of this recommendation is to prevent discharge of steam directly to the drywell, which occurs whenever the safety valves lift. This situation can be avoided if the relief valves (which discharge via piping to an underwater position in the torus) can accommodate all of the necessary excess steam flow.

For NMP-1 the worst pressurization transient is a turbine trip with bypass failure occuring at end-of-cycle. Analysis results[4] provided by the licensee, using the methods described in Reference 8, indicate that because of degrading scram effectiveness power reductions are necessary near and at end-of-cycle in order to maintain a 25 psi pressure margin. However, these initial calculations[4] incorporated conservative nuclear data which resulted in excessive end-of-cycle core power deratings. Accordingly the licensee performed a reanalysis[5] based on updated core nuclear characteristics. The results of these analyses show that a 25 psi margin is available for full power until 1500 Mwd/t prior to EOC-6. However, power limitations of 98% at EOC6-1000 Mwd/t and 95% at EOC6 are required to assure a 25 psi margin. Beginning with the first of the aforementioned exposure points, a power coast-down will be effected until the next lower power level is achieved by fixing control rod position at the start of the exposure interval. Once power falls off to the next lower power level limit, power will be maintained at that value by normal rod motion until the next exposure point is attained. This procedure will then be repeated for the second derate exposure interval.

We find the proposed power coastdown procedure, as described above, to be an acceptable method for assuring the availability of a 25 psi margin to the lowest safety valve setpoint during Cycle 6.

2.7 Power Coastdown Beyond End-of-Cycle

The licensee states[12] that operation beyond the end-of-cycle all rods out condition, in a thermal power coastdown mode, is allowable via reference to the reload topical report[8]. Although our evaluation[9] of the reload topical found the report to be acceptable for reference,we did not specifically include power coastdown operation beyond the end-of-cycle in our review. Accordingly, we do not consider the subject to have been completely addressed generically and cannot find operation in this mode acceptable on a referenced basis.

In response[3] to our request for additional information[7] on this subject, the licensee referenced power coastdown safety analyses[13,14] submitted in connection with similar requests for other operating BWRs. The referenced analyses are for particular BWRs in specific reload cycle core configurations and therefore are not explicitly applicable to Cycle 6 of NMP-1. The referenced analyses show that transient consequences regarding ΔCPR and overpressurization become less severe beyond end-of-cycle. Thus for the same operating limits, margins to core and reactor coolant pressure boundary safety limits increase for burnups beyond the end-of-cycle all rods out condition. The improved transient behavior is predominantly due to the dominant beneficial effect of reduced gross core power level in coastdown operation more than setting the secondary adverse effect of degraded scram reactivity. The analysis assumes a linear rate of power decrease with exposure, which is conservative, since actual thermal power will decrease more rapidly in an exponential manner.

As previously stated, the referenced analyses are not specifically applicable to this plant and cycle. However, we agree with the licensee's argument that the overall trend will be the same for NMP-1 during Cycle 6. Our agreement is restricted to a terminal power level of 70 percent, however. We are confident that down to 70 percent, the scram reactivity insertion rate will not be degraded sufficiently to cause a transient more severe than that of end of cycle. On the above basis we find power coastdown operation, as restricted in a license condition to not less than 70 percent power, to be acceptable. For power coastdown operations to power levels lower than 70 percent, we have requested that cycle and plant-specific analyses or other appropriate justification be provided.

Additionally, neither the current nor proposed Technical Specifications preclude increasing core power level via reduced feedwater heating once operation in the coastdown mode has begun. Such operation, although not planned at this time by the licensee, would negate the assumptions in the referenced analysis as well as the arguments and possibly the conclusions stated above. Accordingly, we require adequate assurance, in the form of a license condition, that feedwater heating capability not be reduced from the normal end-of-cycle operating configuration in order to increase reactor power once into the thermal power coastdown mode.

We have discussed these restrictions with the licensee and he has agreed to these conditions.

3.0 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance the licensee will perform a series of physics startup tests, which are described in Reference 3. Based on our review, this program is acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee within 90 days of startup which is also acceptable.

4.0 Technical Specification Changes

The proposed technical specification changes[5] include a revised fuel cladding integrity safety limit MCPR, revised exposure-dependent operating limit minimum critical power ratios (MCPR) for each fuel type, addition of a MAPLHGR vs. average planar exposure curve and addition of a design maximum total peaking factor for the reload 8x8R fuel assemblies.

The revised 1.07 safety limit MCPR results in a 0.01 increase from the 1.06 safety limit MCPR (SLMCPR) used during Cycle 5. Based on our generic review[7], we find the use of a 1.07 SLMCPR for NMP-1 during Cycle 6 to be acceptable. Also, based on the discussions appearing in Section 2.2.2 herein, the staff finds the proposed operating limit MCPRs to be consistent with and adequately supported by the Reload 7 safety analyses.

The proposed 8x8R design maximum total peaking factor of 3.00 used in connection with the APRM Flux Scram and APRM Rod Block Trip Settings

has been reviewed and found to be acceptable. Additionally, we find the proposed MAPLHGR vs average planar exposure curves for the 8x8R fuel assemblies to be adequate to assure conformance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.

Finally, the current NMP-1 Technical Specifications require that the reactor be brought to the Cold Shutdown condition within 36 hours if any core related thermal parameter,(i.e. APLHGR, LHGR, MCPR or power/flow relationship) which is in violation of its respective operating limit, cannot be returned to within the prescribed limit within two (2) hours. The licensee states, however, that based on previous experience a core power reduction of 10 percent or less is sufficient in most cases to return the parameter to within prescribed limits. Although the violation would be corrected the current technical specifications would require that reactor power reductions be continued and Cold Shutdown conditions achieved. .The proposed technical specifications would require instead that reactor power reductions at a rate not less than 10 percent per hour be initiated if all core related thermal parameters cannot be returned to within prescribed limits within two (2) hours.

Violation of any of the aforementioned core thermal operating limits will not in and of itself cause a degradation of fuel integrity which would necessitate a reactor shutdown and cooldown. We believe that the revised requirements provide for a level of operator action which is commensurate with the safety significance of the observed condition. Accordingly we find the proposed changes to be acceptable.

5.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

6.0 Conclusion

We have concluded that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with

the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 2, 1979

7.0 References

1. Letter to the Director, NRR, from LeBoeuf, Lamb, Leiby and McRae (Counsel for Niagara Mohawk Power Corporation) dated November 21, 1978.

2. Niagara Mohawk Power Corporation letter (Dise) to USNRC (Ippolito) dated January 2, 1979.

3. Niagara Mohawk Power Corporation letter (Schneider) to NRC (Ippolito) dated February 12, 1979.

4. Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1 Reload No. 7, NEDO-24155, November 1978.

5. Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1, Reload No. 7 Reanalysis Supplement, NEDO-24155-1, December 1978.

6. Proposed Changes to Technical Specifications (Appendix A) appearing as Attachment A to the Letter to the Director, NRR, from LeBoeuf, Lamb, Leiby and McRae, dated November 21, 1978.

7. USNRC letter (Ippolito) to Niagara Mohawk Power Corporation (Dise) dated January 19, 1979.

8. "Generic Reload Fuel Application,"NEDE-24011-P-A, May 1977.

9. USNRC letter (Eisenhut) to General Electric (Gridley), dated May 12, 1978, transmitting "Safety Evaluation for the General Electric Topical Report,'Generic Reload Fuel Application,'(NEDO-24011-P)."

10. "Status Report on the Licensing Topical Report General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1; by the Division of Technical Review, Office of Nuclear Reactor Regulation, USNRC April 1975.

11. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360 Revision 1, Supplement 4, April 1, 1976.

12. Attachment B to the letter to the Director, ONRR from LeBoeuf, Lamb, Leiby and McRae, dated November 21, 1978.

13. R. L. Bolger (CECO) letter to B. C. Rusche (NRC), "Quad-Cities Station Unit 2 Proposed Amendment to Facility License No. DPR-30, Docekt No. 50-265," dated June 11, 1976.

14. R. L. Bolger (CECO) letter to E. G. Case (NRC), "Dresden Station Unit 2 Proposed Amendment to Facility Operating License No. DPR-19 to Permit Power Coastdown from 70% Power to 40% Power, NRC Docket No. 50-237," dated June 6, 1977.