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

August 11, 1993

Docket No. 50-410

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M85937)

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2 (NMP-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated February 27, 1993, as supplemented June 18, 1993.

The amendment revises TS Definition 1.9, "Critical Power Ratio," to replace the designation for the General Electric critical power correlation, "GEXL," with a more generic term. Changes have also been made to TS Bases Sections B2.1 and B3/4.2 to reflect the change to Definition 1.9, incorporate revisions to General Electric Company's approved analytical techniques, update references, and reflect changes made to the Reload Section of the NMP-2 Updated Safety Analysis Report.

Copies of the related Safety Evaluation and Notice of Partial Withdrawal are enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice and the Notice of Partial Withdrawal will be published separately in the Federal Register.

Sincerely,

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 46 to NPF-69
2. Safety Evaluation
3. Notice of Partial Withdrawal

cc w/enclosures:
See next page

CP-1

NRC FILE CENTER COPY

160057

DF01

cc:

Mark J. Wetterhahn, Esquire
Winston & Strawn
1400 L Street, NW.
Washington, DC 20005-3502

Mr. Richard Goldsmith
Syracuse University
College of Law
E. I. White Hall Campus
Syracuse, New York 12223

Resident Inspector
Nine Mile Point Nuclear Station
P. O. Box 126
Lycoming, New York 13093

Gary D. Wilson, Esquire
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Mr. David K. Greene
Manager Licensing
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, New York 13126

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Mr. Richard M. Kessel
Chair and Executive Director
State Consumer Protection Board
99 Washington Avenue
Albany, New York 12210

Mr. John H. Mueller
Plant Manager, Unit 2
Nine Mile Point Nuclear Station
Niagara Mohawk Power Corporation
P. O. Box 32
Lycoming, New York 13093

Vice President - Nuclear Generation
Nine Mile Point Nuclear Station
Niagara Mohawk Power Corporation
P. O. Box 32
Lycoming, New York 13093

DATED: August 11, 1993

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-69-NINE MILE POINT UNIT 2

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan (4)
J. Menning
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
Wanda Jones, P-370
C. Grimes, 11/F/23
J. Donoghue, 8/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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. 46
License No. NPF-69

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated February 27, 1993, as supplemented June 18, 1993, 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 1;

   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 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-69 is hereby amended to read as follows:

(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. 46 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 11, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-2	1-2
B2-1	B2-1
B2-2	B2-2
B2-3	B2-3
B2-4	B2-4
B3/4 2-2	B3/4 2-2
B3/4 2-3	B3/4 2-3
B3/4 2-4	B3/4 2-4

CHANNEL FUNCTIONAL TEST

1.6 (Continued)

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps so that the entire channel is tested.

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPs or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the highest value of the FLPD which exists in the core.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of an approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual fuel assembly operating power.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, expressed in microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, expressed in MeV, for isotopes, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump

2.1 BASES FOR SAFETY LIMITS

2.1.0 INTRODUCTION

The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set so that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit so that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses that occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. Although fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions that would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of critical power correlations is not valid for all critical power calculations performed at reduced pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5-psi driving head will be greater than 28×10^3 lb/hr. Full-scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set so that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an approved critical power correlation. The critical power correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2. The bases for the uncertainties in the core parameters and the basis for the uncertainty in the critical power correlation are given in Reference 1. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

References:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

BASES TABLE B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT*

QUANTITY	STANDARD DEVIATION (% OF POINT)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
R Factor	1.6
Critical Power	3.0

* The uncertainty analysis used to establish the corewide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core. The values herein apply to both two recirculation loop operation and single recirculation loop operation.

BASES TABLE B2.1.2-2

NOMINAL VALUES OF PARAMETERS* USED IN THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

PARAMETER	VALUE
THERMAL POWER	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1005 psig
Bundle Enrichment	3.0 Wt % U-235
R-Factor:	
0 - 10 GWD/ST	0.915
10 - 15 GWD/ST	0.954
> 15 GWD/ST	0.954

* The values in this table are for a representative plant.

BASES

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient, assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The evaluation of a given transient begins with the system initial parameters shown in USAR Tables 15.0-3 and A15.0-4 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2. The principal result of this evaluation is the reduction in MCPR caused by the transients.

The purpose of the K_f factor specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of operating limit MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated. The K_f factors are calculated as described in Reference 2.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, an MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts,

BASES TABLE B3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS*

PARAMETERS	VALUE
Plant:	
1. Core THERMAL POWER	3461 MWt** which corresponds to 105% of rated steam flow
2. Vessel Steam Output	15.0×10^6 lbm/hr which corresponds to 105% of rated steam flow
3. Vessel Steam Dome Pressure	1055 psia
4. Design Basis Recirculation Line Break Area for:	
a. Large Breaks	3.1 ft^2
b. Small Breaks	0.09 ft^2

Fuel:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO†
Initial Core	8 x 8	13.4	1.4	1.20
Reload	8 x 8	14.4	1.4	1.20

* A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the USAR.

** This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

† For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 second after LOCA regardless of initial MCPR.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

3/4.2.3 (Continued)

while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures MCPR will be known following a change in THERMAL POWER or power shape, and therefore avoid operation while exceeding a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the linear heat generation rate (LHGR) in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation exceeding a thermal limit.

References

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, latest approved revision.

2. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, latest approved revision.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated February 27, 1993 (Reference 1), as supplemented June 18, 1993 (Reference 2), Niagara Mohawk Power Corporation (NMPC or the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station, Unit 2 (NMP-2), Technical Specifications (TSs). The requested changes would modify the recirculation flow upscale rod block setpoint and permit the use of NRC-approved power correlations other than the GEXL correlation for NMP-2 reload analyses. GEXL is the term used to refer to a specific NRC-approved power correlation (GE Critical Quality, X_c, Boiling Length, L_b) developed to predict the onset of transition boiling. Changes to the TS Bases were also proposed that would reflect the use of NRC-approved power correlations, incorporate revisions to General Electric Company's (GE's) approved analytical techniques, update references, and reflect changes made to the Reload Section of the NMP-2 Updated Safety Analysis Report (USAR).

The June 18, 1993, letter withdrew the proposed changes to the recirculation flow upscale rod block setpoint. These changes had been proposed to facilitate operation at up to 105% of rated core flow. The licensee stated that this rod block performs no safety function and no design basis transient or accident analysis takes credit for it. That is, there is no safety basis for the setpoint. The licensee stated that the recirculation flow upscale rod block changes were being withdrawn since they could not at that time provide the staff with an analytical basis for the proposed changes. The licensee further stated that they would evaluate the necessity of retaining the recirculation flow rod blocks in the TSs and may propose removal of these rod blocks in a future amendment application. The licensee's letter of June 18, 1993, did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Critical power ratio (CPR) is defined in TS Definition 1.9. CPR is the power in a fuel assembly that causes some point in the assembly to experience boiling transition divided by the actual fuel assembly power. Definition 1.9 currently specifies that the boiling transition power level is to be determined using the GEXL correlation.

As GE has further developed its analytical techniques, critical power correlations other than GEXL have been used in boiling water reactor loss-of-coolant accident analyses and approved by the NRC. For instance, use of the GEXL-PLUS correlation was approved in Reference 3. The proposed change to Definition 1.9 is administrative in nature and would allow the use of any NRC-approved critical power correlation to determine operating limit minimum CPRs. The staff finds this proposed change to be acceptable.

The licensee has proposed changes to the TS Bases that are listed below. These changes reflect updated analytical techniques and new fuel information furnished by GE, modifications to the USAR, and the proposed change to Definition 1.9.

a. A reference to the GEXL correlation in Bases paragraph 2.1.1 would be replaced with a general reference to critical power correlations.

b. The discussion in Bases paragraph 2.1.2 would be revised to remove obsolete references, as well as a reference to the GEXL correlation, and incorporate a reference to NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

c. Bases Tables B2.1.2-1 and B2.1.2-2 would be updated to reflect new fuel (GE9) values and to replace plant specific data with nominal parameters for fuel integrity analysis.

d. References in Bases Section 3/4.2.3 would be revised to include the reload analysis section of the USAR (Appendix A), and to remove obsolete references.

e. Bases Table B3.2.1-1 would be revised to include new fuel information.

Some of the proposed Bases changes are required to reflect the revision of TS Definition 1.9. The other proposed Bases changes are supported by analytical techniques or new fuel design information previously approved by the NRC. Therefore, the staff has no objections to these proposed changes.

By letter dated June 18, 1993, NMPC requested a withdrawal of a portion of the proposed changes. The licensee requested that the proposed changes to the recirculation flow upscale rod block setpoint be withdrawn, since they could not at that time provide the staff with an analytical basis for the proposed changes. The staff finds this acceptable. A Notice of Partial Withdrawal will be published in the Federal Register.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 16866). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 References

1. Letter (NMP2L 1370) from B. Ralph Sylvia (NMPC) to USNRC, dated February 27, 1993.

2. Letter (NMP2L 1393) from B. Ralph Sylvia (NMPC) to USNRC, dated June 18, 1993.

3. Letter from A. S. Thadani (USNRC) to J. S. Charnley (GE), dated March 14, 1988.

Principal Contributor:
Joseph Donoghue

Date: August 11, 1993

UNITED STATES NUCLEAR REGULATORY COMMISSION

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT 2

NOTICE OF PARTIAL WITHDRAWAL OF APPLICATION FOR

AMENDMENT TO FACILITY OPERATING LICENSE

The United States Nuclear Regulatory Commission (the Commission) has granted the request by Niagara Mohawk Power Corporation (NMPC) to withdraw a portion of their February 27, 1993, application for a proposed amendment to Facility Operating License NPF-69 for Nine Mile Point Nuclear Station, Unit 2, located in Oswego County, New York.

The proposed amendment involved changes to the Technical Specifications (TSs) to modify the recirculation flow upscale rod block setpoint and permit the use of NRC-approved power correlations other than the GEXL correlation. Changes to the TS Bases were also proposed that would reflect the use of NRC-approved power correlations, incorporate revisions to General Electric Company's approved analytical techniques, update references, and reflect changes made to the Reload Section of the Updated Safety Analysis Report.

On June 18, 1993, the licensee submitted a letter to the NRC requesting withdrawal of the proposed change to the recirculation flow upscale rod block setpoint. The licensee requested withdrawal since it could not at that time provide the Commission with an analytical basis for the proposed change. The recirculation flow upscale rod block performs no safety function and no design basis transient or accident analysis takes credit for it.

The Commission has previously issued a Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing which was published in the FEDERAL REGISTER on March 31, 1993 (58 FR 16866).

For further details with respect to this action, see the application for amendment dated February 27, 1993, and the licensee's letter of June 18, 1993, which withdrew the portion of the application for license amendment. The above documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC 20555 and at the Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York.

Dated at Rockville, Maryland, this 11th day of August 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-410

August 11, 1993

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M85937)

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2 (NMP-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated February 27, 1993, as supplemented June 18, 1993.

The amendment revises TS Definition 1.9, "Critical Power Ratio," to replace the designation for the General Electric critical power correlation, "GEXL," with a more generic term. Changes have also been made to TS Bases Sections B2.1 and B3/4.2 to reflect the change to Definition 1.9, incorporate revisions to General Electric Company's approved analytical techniques, update references, and reflect changes made to the Reload Section of the NMP-2 Updated Safety Analysis Report.

Copies of the related Safety Evaluation and Notice of Partial Withdrawal are enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice and the Notice of Partial Withdrawal will be published separately in the Federal Register.

Sincerely,
Original signed by:

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 46 to NPF-69
2. Safety Evaluation
3. Notice of Partial Withdrawal

cc w/enclosures:
See next page

Distribution:
See attached sheet

LA:PDI-1	PM:PDI-1	OTSB	OGC	D:PDI-1	
CVogan	JMenning:smm	CGrimes		RACapra	
7/9/93	7/9/93	7/12/93	8/5/93	8/11/93	/ /

OFFICIAL DOCUMENT COPY
FILENAME: NM285937.AMD