April 19, 1988

Docket No. 50-220

DISTRIBUTION
Docket File
NRCPDR
Local PDR
PDI-1 Rdg.
CVogan
RBenedict
OGC
SVarga
BBoger
JJohnson,RI
EJordan
JPartlow
Wanda Jones
DHagan
ACRS(10)
TBarnhart(4)
GPA/PA
ARM/LFMB
EButcher
THuang

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

Dear Mr. Mangan:

The Commission has issued the enclosed Amendment No. 97 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit 1 (NMP-1). The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated August 21, September 14, December 17, and December 18, 1987; and as supplemented March 9, 1988 (TACS 66552 and 66968).

This amendment revises portions of Technical Specifications 2.1.1 and 3.1.7 and their Bases to reflect new methodology in establishing Maximum Average Planar Linear Heat Generation Rates (MAPLHGR); to reflect the Maximum Total Peaking Factor and MAPLHGR limits for a new fuel type; and to change exposure-dependent Minimum Critical Power Ratio (MCPR) limits to one limit that is applicable to the entire cycle.

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

Sincerely,

Robert A. Benedict, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 97 to DPR-63
2. Safety Evaluation

cc: w/enclosures
See next page

* SEE PREVIOUS CONCURRENCE

PDI-1	RAB PDI-1	OGC	RC PDI-1
CVogan*	RBenedict*:mak	MYoung*	RCapra
3/23/88	3/23/88 4/18/88	3/25/88	4/19/88

Docket No. 50-220

DISTRIBUTION

Docket File	EJordan
NRCPDR	JPartlow
Local PDR	Wanda Jones
PDI-1 Rdg.	DHagan
CVogan	ACRS(10)
RBenedict	TBarnhart(4)
OGC	GPA/PA
SVarga	ARM/LFMB
BBoger	EButcher
JJohnson,RI	THuang

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

Dear Mr. Mangan:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit 1 (NMP-1). The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated August 21, September 14, December 17 and December 18, 1987, and March 9, 1988 (TACS 66552 and 66968).

This amendment revises portions of Technical Specifications 2.1.1 and 3.1.7 and their Bases to reflect new methodology in establishing Maximum Average Planar Linear Heat Generation Rates (MAPLHGR); to reflect the Maximum Total Peaking Factor and MAPLHGR limits for a new fuel type; and to change exposure-dependent Minimum Critical Power Ratio (MCPR) limits to one limit that is applicable to the entire cycle.

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

Sincerely,

Robert A. Benedict, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. to DPR-63
2. Safety Evaluation

cc: w/enclosures
See next page

PDI-1	RAB PDI-1	OGC MY w/ comments noted to SE, ltr & notice	PDI-1
CVogan	RBenedict:mak	MYoung	RCapra
3/23/88	3/23/88	3/25/88	3/ /88

Mr. C. V. Mangan
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station,
Unit No. 1

cc:

Troy B. Conner, Jr., Esquire
Conner & Wetterhahn
Suite 1050
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Frank R. Church, Supervisor
Town of Scriba
R. D. #2
Oswego, New York 13126

Mr. Thomas Perkins
General Supt.-Nuclear Generation
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Post Office Box 32
Lycoming, New York 13093

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 126
Lycoming, New York 13093

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

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

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

Mr. Thomas W. Roman
Unit 1 Station Superintendent
Nine Mile Point Nuclear Station
Post Office Box 32
Lycoming, New York 13093

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. DPR-63

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

   A. The applications for amendment by Niagara Mohawk Power Corporation of New York, Inc. (the licensee) dated August 21, September 14, December 17, and December 18, 1987; and as supplemented March 9, 1988, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 97 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 19, 1988

ATTACHMENT TO LICENSE AMENDMENT

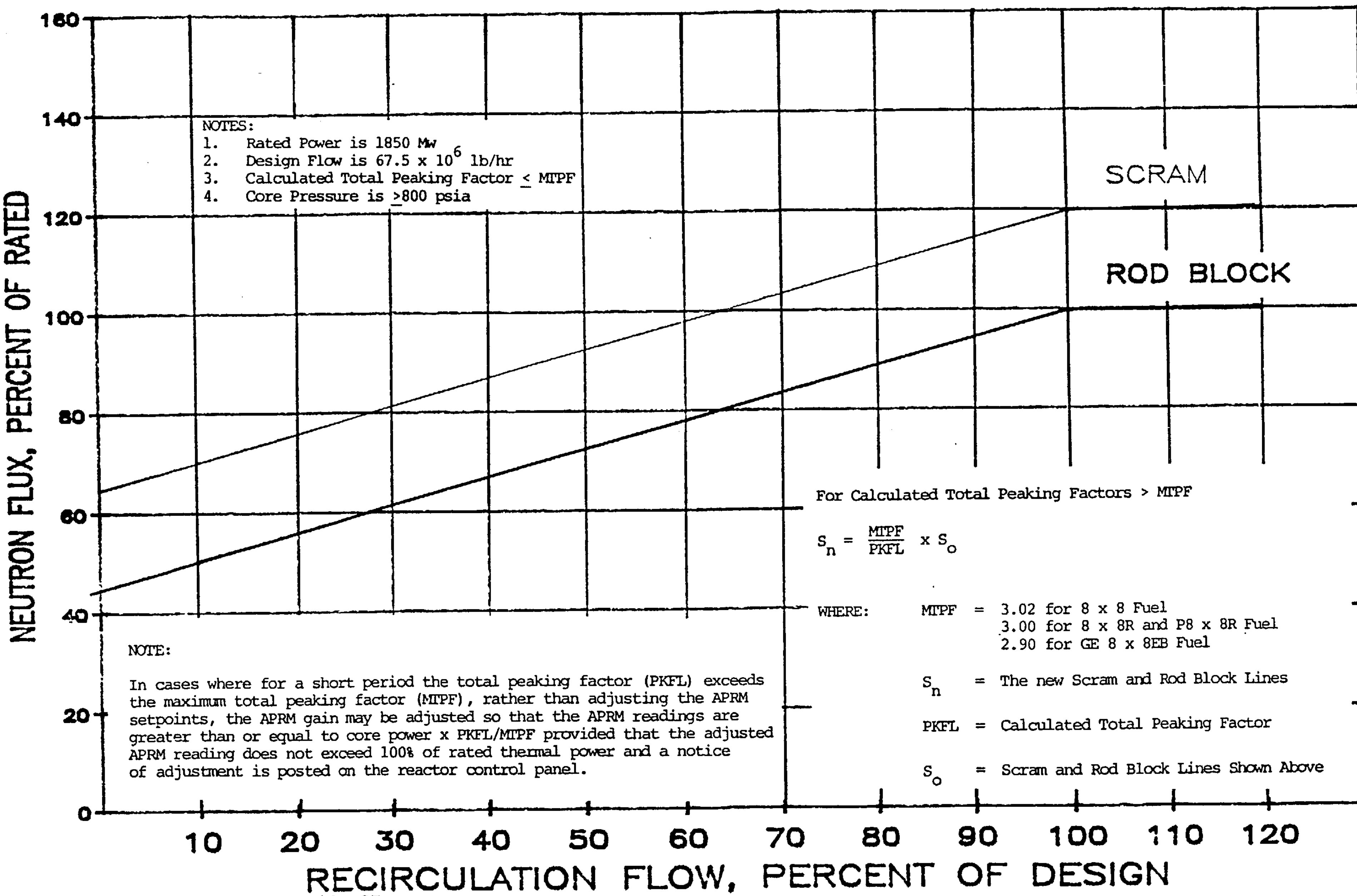
AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A as follows:

Remove Pages	Insert Pages
8	8
11	11
20	20
63	63
64a	64a
64b	64b
64c	64c
69a	69a
--	69a1
70	70
70b	70b
70d	70d

FIGURE 2.1.1
FLOW BIASED SCRAM AND APRM ROD BLOCK



Amendment No. ~~31~~, ~~41~~, ~~92~~, 97

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of the SLCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit SLCPR operation is constrained to a maximum LHGR of 13.4 kW/ft for 8x8, 8x8R, P8x8R and GE8x8EB fuel (Reference 15). At 100% power, this limit is reached with a Maximum Total Peaking Factor (MTPF) of 3.02 for 8x8 fuel, 3.00 for 8x8R and P8x8R fuel, and 2.90 for GE8x8EB fuel. During steady-state operation where the total peaking factor is above 2.90, the equation in Figure 2.1.1 will be used to adjust the flow biased scram and APRM rod block set points.

At pressure equal to or below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all core flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 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. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28×10^3 lb/hr

REFERENCES FOR BASES 2.1.1 AND 2.1.2 FUEL CLADDING

(1) General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO-10958 and NEDE-10958.

(2) Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, February 1973.

(3) FSAR, Volume II, Appendix E.

(4) FSAR, Second Supplement.

(5) FSAR, Volume II, Appendix E.

(6) FSAR, Second Supplement.

(7) Letters, Peter A. Morris, Director of Reactor Licensing, USAEC, to John E. Logan, Vice-President, Jersey Central Power and Light Company, dated November 22, 1967 and January 9, 1968.

(8) Technical Supplement to Petition to Increase Power Level, dated April 1970.

(9) Letter, T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.

(10) Letter, Philip D. Raymond, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, dated October 15, 1973.

(11) Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis, NEDO 24012, May, 1977.

(12) Licensing Topical Report General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, May, 1986.

(13) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(14) General Electric SIL 299 "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation."

(15) Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-B, Amendment 10."

3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, 3.1.7e, 3.1.7f and 3.1.7g. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION

c. Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR for all 8 x 8 fuel at rated power and flow shall be as shown in the table below:

LIMITING CONDITION FOR OPERATION MCPR

Core Average Incremental Exposure	Limiting MCPR*
BOC to EOC	$\geq$ 1.40

If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit.

For core flows other than rated the MCPR limit shall be the limit identified above times K_f where K_f is as shown in Figure 3.1.7-1.

d. Power Flow Relationship During Operation

The power/flow relationship shall not exceed the limiting values shown in Figure 3.1.7.aa.

* This limit shall be determined to be applicable each operating cycle by analyses performed utilizing the ODYN transient code.

SURVEILLANCE REQUIREMENT

c. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at > 25% rated thermal power.

d. Power Flow Relationship

Compliance with the power flow relationship in Section 3.1.7.d shall be determined daily during reactor operation.

e. Partial Loop Operation

Under partial loop operation, surveillance requirements 4.1.7.a,b,c and d above are applicable.

LIMITING CONDITION FOR OPERATION

If at any time during power operation, it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until the power/flow relationship is within the prescribed limits.

e. Partial Loop Operation

During power operation, partial loop operation is permitted provided the following conditions are met.

When operating with four recirculation loops in operation and the remaining loop unisolated, the reactor may operate at 100 percent of full licensed power level in accordance with Figure 3.1.7aa and an APLHGR not to exceed 98 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e and an APLHGR not to exceed 99% of the limiting values shown in Figures 3.1.7f and 3.1.7g.

When operating with four recirculation loops in operation and one loop isolated, the reactor may operate at 100 percent of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed 98 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e and an APLHGR not to exceed 99% of the limiting values shown in Figures 3.1.7f and 3.1.7g, provided the following conditions are met for the isolated loop.

1. Suction valve, discharge valve and discharge bypass valve in the isolated loop shall be in the closed position and the associated motor breakers shall be locked in the open position.

SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

2. Associated pump motor circuit breaker shall be opened and the breaker removed.

If these conditions are not met, core power shall be restricted to 90.5 percent of full licensed power.

When operating with three recirculation loops in operation and the two remaining loops isolated or unisolated, the reactor may operate at 90% of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed 96 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e and an APLHGR not to exceed 99% of the limiting values shown in Figures 3.1.7f and 3.1.7g.

During 3 loop operation, the limiting MCPR shall be increased by 0.01.

Power operation is not permitted with less than three recirculation loops in operation.

If at any time during power operation, it is determined by normal surveillance that the limiting value for APLHGR under one and two isolated loop operation is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is not returned to within the prescribed limits for one and two isolated loop operation within two (2) hours, reactor power reduction shall be initiated at a rate not less than 10 percent per hour until APLHGR at all nodes is within the prescribed limits.

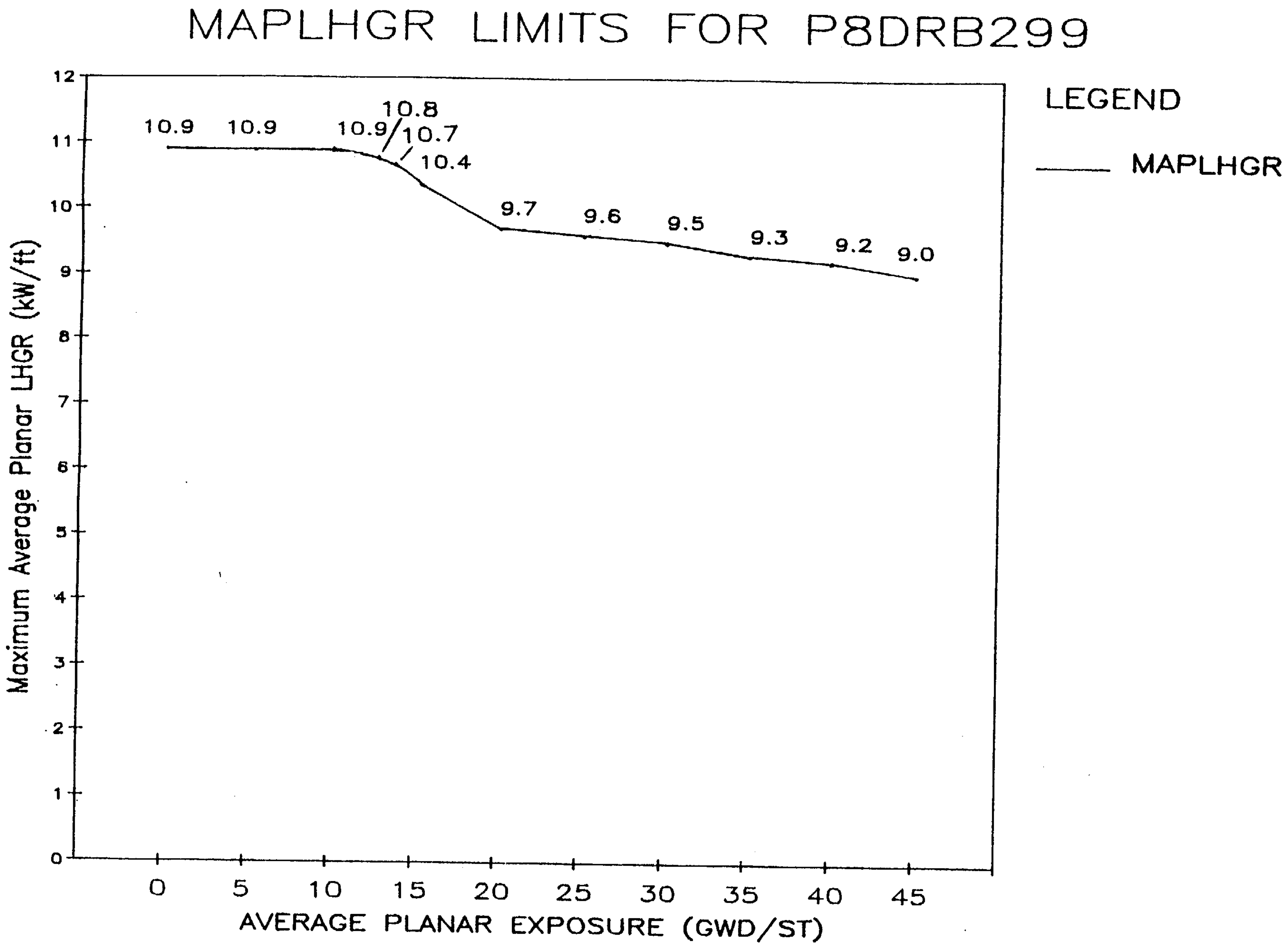


Figure 3.1.7f Maximum Allowable Average Planar LHGR Applicable to P8DRB299 and Future Reload Fuel as described in Reference 15.

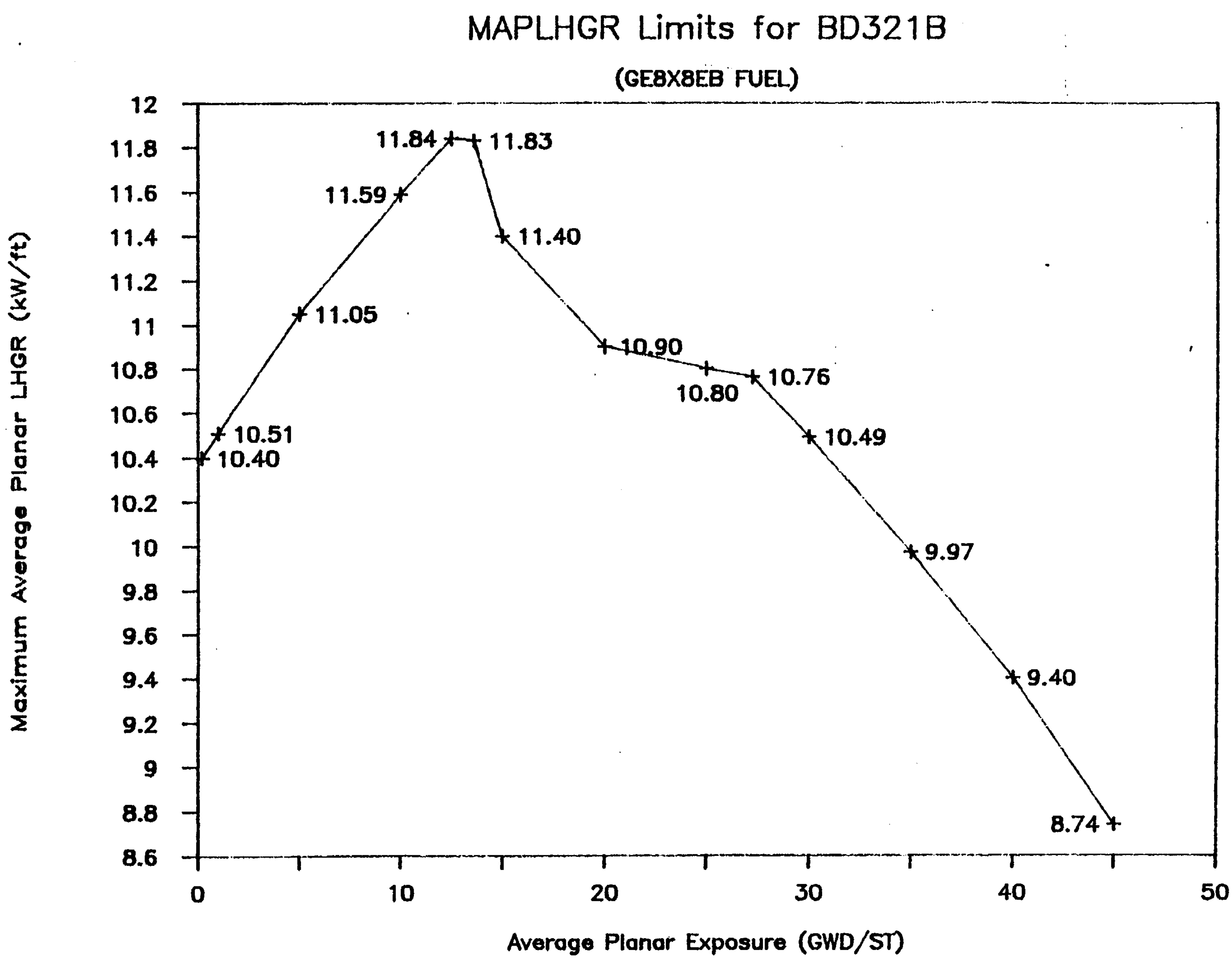


Figure 3.1.7g Maximum Allowable Average Planar LHGR Applicable to BD321B and Future Reload Fuel as described in Reference 16

Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature and the peak local cladding oxidation following the postulated design basis loss-of-coolant accident will not exceed the limits specified in 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20 F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is shown in Figures 3.1.7a-g. These curves are based on calculations using the models described in References 1, 2, 3, 5, 6, 13, 15 and 16.

The Reference 13 and 15 LOCA analyses are sensitive to minimum critical power ratio (MCPR). In the Reference 15, analysis a MCPR value of 1.30 was assumed. If future transient analyses should yield a MCPR limit below this value, the Reference 15 LOCA analysis MCPR value would become limiting. The current MCPR limit is ≥ 1.40. For fuel bundles analyzed with the Reference 13 LOCA methodology, assume MCPR values of 1.30 and 1.36 for five recirculation loop and less than five loop operation respectively.

Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated (Reference 12). The LHGR shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at a 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 and thermal-hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing

Partial Loop Operation

The requirements of Specification 3.1.7e for partial loop operation in which the idle loop is isolated, precludes the inadvertent startup of a recirculation pump with a cold leg. However, if these conditions cannot be met, power level is restricted to 90.5 percent power based on current transient analysis (Reference 9). For three loop operation, power level is restricted to 90 percent power based on the Reference 13 and 15 LOCA analyses.

The results of the ECCS calculation are affected by one or more recirculation loops being unisolated and out of service. This is due to the fact that credit is taken for extended nucleate boiling caused by flow coastdown in the unbroken loops. The reduced core flow coastdown following the break results in higher peak clad temperature due to an earlier boiling transition time. The results of the ECCS calculations are also affected by one or more recirculation loops being isolated and out of service. The mass of water in the isolated loops unavailable during blowdown results in an earlier uncovery time for the hot node. This results is an increase in the peak clad temperature.

For fuel bundles analyzed with the methodology used in Reference 13, MAPLHGR shall be reduced 2% and 4% for 4 and 3 loop operation respectively. For fuel bundles analyzed with the methodology used in References 15 and 16, MAPLHGR shall be reduced by 1% for both 4 and 3 loop operation.

Partial loop operation and its effect on lower plenum flow distribution is summarized in Reference 11. Since the lower plenum hydraulic design in a non-jet pump reactor is virtually identical to a jet pump reactor, application of these results is justified. Additionally, non-jet pump plants contain a cylindrical baffle plate which surrounds the guide tubes and distributes the impinging water jet and forces flow in a circumferential direction around the outside of the baffle.

Recirculation Loops

Requiring the suction and discharge for at least two (2) recirculation loops to be fully open assures that an adequate flow path exists from the annular region between the pressure vessel wall and the core shroud, to the core region. This provides for communication between those areas, thus assuring that reactor water level instrument readings are indicative of the water level in the core region.

When the reactor vessel is flooded to the level of the main steam line nozzle, communication between the core region and annulus exists above the core to ensure that indicative water level monitoring in the core region exists. When the steam separators and dryer are removed, safety limit 2.1.1d and e requires water level to be higher than 9 feet below minimum normal water level (Elevation 302'9"). This level is above the core shroud elevation which would ensure communication between the core region and annulus thus ensuring indicative water level monitoring in the core region. Therefore, maintaining a recirculation loop in the full open position in these two instances are not necessary to ensure indicative water level monitoring.

(1) "Fuel Densification Effects on GE Boiling Water Reactor Fuel," Supplements 6, 7 and 8, NEDM-10735, August 1973.

(2) Supplement 1 to Technical Report on Densifications of GE Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

(3) Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

(4) "GE Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Supplement 1 to Rev. 1, December 1974.

(5) GE Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K," NEDO-20566.

(6) GE Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G.L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

(7) "Nine Mile Point Nuclear Power Station Unit 1, Load Line Limit Analysis," NEDO-24012.

(8) Licensing Topical Report GE Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, August 1978.

(9) Final Safety Analysis Report, Nine Mile Point Nuclear Station, Niagara Mohawk Power Corporation, June 1967.

(10) NRC Safety Evaluation, Amendment No. 24 to DPR-63 contained in letter from G. Lear, NRC, to D. P. Dise dated May 15, 1978.

(11) "Core Flow Distribution in a GE Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A.

(12) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(13) Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit 1 Nuclear Power Station, NEDO-24348, Aug. 1981.

(14) GE Boiling Water Reactor Extended Load Line Limit Analysis for Nine Mile Point Unit 1 Cycle 9, NEDC-31126, February 1986.

(15) Nine Mile Point Unit 1, Loss-of-Coolant Accident Analysis, NEDC-31446P, June 1987.

(16) Supplement 1 to Nine Mile Point Generating Station Unit 1 SAFER/CORECOOL/GESTR-LOCA Analysis Report NEDC-31446P-1, Class III, September 1987.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-220

INTRODUCTION

By letter (NMP1L 0177) from C. V. Mangan, Niagara Mohawk Power Corporation (NMPC), to NRC dated August 21, 1987 (Ref. 1), Technical Specification (TS) changes were requested for Specification 3.1.7, Figure 3.1.7f and the Bases for 3.1.7 and 4.1.7 set forth in Appendix A to that license be amended to reflect new methodology in establishing the Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) for the P8DRB299 fuel type. In a separate submittal by letter (NMP1L 0210) from T. Lempges (NMPC) to NRC dated December 18, 1987 (Ref. 2), NMPC has proposed that Specifications 2.1.1 and 3.1.7, Figures 2.1.1 and 3.1.7g, and the associated Bases for 2.1.1, 3.1.7, and 4.1.7 be amended in order to reflect the Maximum Total Peaking Factor and addition of the MAPLHGR for the General Electric Fuel bundle type BD321B (GE8x8EB) (Ref. 2a).

By letter (NMP1L 1086) dated September 14, 1987, the licensee applied for withholding from public disclosure, as proprietary, a report that accompanied the August 21, 1987 letter. A non-proprietary version of that report was provided with the licensee's letter NMP1L 0208, dated December 17, 1987. The staff's consideration of the September 14 and December 17 letters has only to do with making a finding related to the proprietary nature of a document and does not affect this safety evaluation; it will be reported separately.

By letter (NMP1L 0232) dated March 9, 1988, the licensee provided clarifying information concerning the new fuel and indicated a minor change in the fuel mix for Cycle 10. The effect of this change has been considered in the staff's evaluation. Because the submittal provided supplemental information which did not modify any proposed TS, it did not affect the substance of the proposed action or the staff's initial determination published in the Federal Register on February 10, 1988.

DISCUSSION AND EVALUATION

Reload Description

The Nine Mile Point Unit 1 (NMP-1) Cycle 10 reload (Ref. 3) will retain 156 P8DNB277 fuel assemblies from Cycle 8 and 200 P8DRB299 fuel assemblies from Cycle 9, and will add 176 new BD321B fuel assemblies (GE8x8EB) (Ref. 2a). The loading will be a conventional scatter pattern with low reactivity fuel on the periphery.

Fuel Design

The new fuel assembly to be used for NMP-1 Cycle 10, BD321B (GE8x8EB fuel) has been approved for inclusion in NEDE-24011, GESTAR II (Amendment 18). This fuel type has been analyzed for this application (Refs. 4a and 4b) with approved methods (Ref. 5) and meets the approved limits of GESTAR II (Ref. 6). Therefore, the new fuel is acceptable for NMP-1 Cycle 10.

Nuclear Design

The nuclear design for NMP-1 Cycle 10 has been performed with the methodology described in GESTAR II (Ref. 6). The results of those analyses are given in Reference 3. The shutdown margin (SDM) is 4.6% delta k at the beginning of cycle and 1.2% delta k at the minimum conditions. Therefore, it meets the required .38% delta k shutdown margin. The standby liquid control system also meets shutdown requirements with a shutdown margin of 4.1% delta k. Since these and other NMP-1 Cycle 10 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

Thermal-Hydraulic Design

The thermal-hydraulic design for NMP-1 Cycle 10 has been performed with the methodology described in GESTAR II (Ref. 6) and the results are given in Reference 3 for the NMP-1.

The licensee has proposed that two MAPLHGR curves for the fresh fuel bundles of BD321B (Figure 3.1.7g) and P8DRB299 (Figure 3.1.7f) be added to the NMP-1 Cycle 10 Technical Specifications. These MAPLHGR curves are generated based on the approved methodology (Ref. 5) and the results, which conform to 10 CFR 50 Appendix K requirements and meet 10 CFR 50.46 criteria, are given in References 4a and 4b. We find these changes are acceptable.

The licensee has also proposed to eliminate exposure dependent Minimum Critical Power Ratio (MCPR) limits and to use one MCPR which is applicable for the entire cycle. The MCPR limits were calculated using approved methodology (Ref. 6) and documented in Reference 3. The limiting transients have been analyzed and the results indicate that if a MCPR of 1.37 is maintained throughout the cycle, it will assure that the safety limit MCPR will not go below 1.07. Therefore, we find the TS MCPR of 1.4 through the entire cycle to be acceptable.

NMP-1 Cycle 10 uses the approved GE fuel type GE8x8EB which has been shown to have adequate stability margin (Ref. 7) and therefore is acceptable and its reload cycle is exempted from the current requirement to submit a cycle specific stability analysis to the NRC.

Transient and Accident Analyses

The transient and accident analysis methodologies used for NMP-1 Cycle 10 are described in GESTAR II (Ref. 6) and the results are provided in Reference 3. The core wide transient including loss of 100°F feedwater heating, turbine trip without bypass and feedwater controller failure, local rod withdrawal error, and the Main Steam Isolation Valve Closure (no scram) are performed using approved methods (Ref. 6) and the results are acceptable and fall within expected ranges.

The Rod Drop Accident (RDA) was not specifically analyzed for NMP-1 Cycle 10. NMP-1 uses a Banked Position Withdrawal Sequence for control rod withdrawal. For plants using this system the RDA event has been statistically analysed generically and it was found that with a high degree of confidence the peak fuel enthalpy would not approach the NRC limit of 280 cal/gm for this event. This approach and analysis has been approved by the NRC (Ref. 6). This approach is acceptable for NMP-1 Cycle 10.

Technical Specifications

The Technical Specification changes are for the most part minor and provide the MAPLHGR limits for a new fuel type. Details of the specification changes follow:

1) Specification 2.1.1, Bases 2.2.1 and Figure 2.1.1 - Changes include the formula contained in Figure 2.1.1 for adjusting the flow biased scram and APRM rod block setpoints in those cases where the calculated total peaking factor exceeds the maximum total peaking factor for the fuel type, specifically the GE8x8EB is the new fuel to be added to the core during the 1988 refueling and maintenance outage. The maximum total peaking factor for the GE8x8EB fuel was calculated by GE to be 2.90. This change has been clarified in the note of Figure 2.1.1, which reads: in cases where for a short period the total peaking factor (PKFL) exceeds the maximum total peaking factor (MTPF), rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that the APRM readings are greater than or equal to core power X PKFL/MTPF provided that the adjusted APRM reading does not exceed 100% of rated thermal power and a notice of adjustment is posted on the reactor control panel. We find that this revision provides needed flexibility during startup and power escalation to rated conditions and is acceptable. Due to addition of new fuel GE8x8EB, a 2.9 maximum total peaking factor (MTPF) for GE8x8EB fuel was included in Bases 2.1.1. We find this to be acceptable. Addition of Reference 15 to References for Bases 2.1.1 and 2.1.2 is acceptable. This Reference 15 is a letter from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Amendment 10."

2) Specification 3.1.7, Bases 3.1.7 and 4.1.7 and Figures 3.1.7f and 3.1.7g - The proposed changes to Specification 3.1.7 and the addition of Figures 3.1.7f and 3.1.7g reflects the use of the SAFER/CORECOOL/GESTAR-LOCA computer codes and methodology (Ref. 4b) and the addition of maximum average planar linear heat generation rate (MAPLHGR) limits for the GE8x8EB fuel. The methods used to analyze the loss of coolant accident response of P8DRB299 and GE8x8EB fuel conform to 10 CFR 50 Appendix K requirement and were approved by the staff (Ref. 5). Therefore, the changes are acceptable. The results of the limiting transients (Ref. 3) indicate that if a minimum critical power ratio of 1.37 is maintained throughout the cycle, it will assure that the minimum critical power ratio will not go below 1.07 during the most limiting transient. The

proposed TS change to 1.40 MCPR throughout the entire fuel cycle is above the minimum required critical power ratio of 1.37. Therefore, the TS change is acceptable.

The supporting documents (refs. 4a and 4b) to be added as references 15 and 16 for Bases 3.1.7 and 4.1.7 are acceptable.

As a result of our review, we conclude that the proposed reload and technical specification changes are acceptable.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

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

Dated: April 19, 1988

PRINCIPAL CONTRIBUTOR:

T. Huang

REFERENCES

1. Letter from C. V. Mangan, (NMPC), to NRC (NMP1L 0177) dated August 21, 1987 with Attachment A, "Proposed Changes to Technical Specifications (Appendix A)," and Attachment B, "Supporting Information and No Significant Hazards Conditions Analysis."

2. Letter from T. E. Lempges, (NMPC), to NRC (NMP1L 0210) dated December 18, 1987 with Attachment A, "Proposed Changes to Technical Specifications (Appendix A)," and Attachment B, "Supporting Information and No Significant Hazards Consideration Analysis."

2a. Letter from C. V. Mangan, (NMPC), to NRC (NMP1L 0232) dated March 9, 1988 with Attachment 23A5862, pages 7 and 14.

3. 23A5862, Revision 0, October 1987, "Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1, Reload 11."

4a. NEDC-31446P, June 1987, "Nine Mile Point Nuclear Station Unit One SAFER/COREC0OL/GESTAR-LOCA Loss-of-Coolant Accident Analysis."

4b. NEDC-31446P-1, September 1987, "Supplement 1 to Nine Mile Point Unit 1 SAFER/CORECOOL/GESTAR-LOCA Loss-of-Coolant Accident Analysis."

5. Letter, A. C. Thadani (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-30996-P, Volume II, 'SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet and Non-jet Pump Plants,'" May 1987.

6. NEDE-24011-P-A-US, May 1986, "General Electric Standard Application for Reactor Fuel," (GESTAR II).

7. Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8 'Thermal Hydraulic Stability Amendment to GESTAR II,'" dated April 24, 1985.