Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

April 28, 1995

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M87088)

Dear Mr. Sylvia:

The Commission has issued the enclosed Amendment No. 66 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 July 22, 1993, as supplemented by letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995.

The amendment modifies Facility Operating License No. NPF-69 and the NMP-2 TSs to authorize an increase in the maximum power level of NMP-2 from 3323 megawatts thermal (MWt) to 3467 MWt. The amendment also approves changes to the TSs to implement uprated power operation.

The amendment is effective as of its date of issuance and is to be implemented prior to restart from refueling outage 4.

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

Sincerely,

Fresh 83852-104

Original signed by:
Gordon E. Edison, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures: 1. Amendment No. 66 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

*See previous concurrence

DOCUMENT NAME: NM287088.AMD

OFFICE	LA:PDI-1	PM:PDI-1	BC:RHFB for	BC:SRXB *	BC:TERB *
NAME	SLittle	GEdison:cn	~~MWohlson~~ CThomas	RJones	CMiller
DATE	04/13/95	04/13/95	04/13/95	04/14/95	04/17/95
OFFICE	BC:EMEB *	BC:SPLB *	BC:SCSB *	BC:HICB *	OGC
NAME	RWessman	CMcCracken	RBarrett	JWermiel	
DATE	04/ /95	04/ /95	04/12/95	04/13/95	04/ /95
OFFICE	D:PDI-1	D:DRPE	ADP:NRR	D:NRR	
NAME	LMarsh	SVarga	RZimmerman	WRussell	
DATE	04/21/95	04/ /95	04/27/95	04/28/95	

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(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. 66 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 and is to be implemented prior to startup from refueling outage 4.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3 and 5 of License*
2. Changes to the Technical Specifications

Date of Issuance: April 28, 1995

*Pages 3 and 5 are attached, for convenience, for the composite license to reflect this change.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 66 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 ated July 22, 1993 (Reference 1), as supplemented by letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995, Niagara Mohawk Power Corporation (the licensee or NMPC) submitted a request for changes to the Nine Mile Point Nuclear Station, Unit 2 (NMP-2), Facility Operating License (NPF-69) and for changes to the NMP-2 Technical Specifications (TSs). The request would increase the licensed thermal power level of the NMP-2 reactor from the current limit of 3323 megawatts thermal (MWt) to 3467 MWt. The request would also approve changes to the TSs to implement uprated power operation. This request is in accordance with the generic boiling-water reactor (BWR) power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter dated September 30, 1991 (Reference 2). NMPC's letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

On December 28, 1990, GE submitted GE Licensing Topical Report (LTR) NEDC-31897P-1, in which it proposed to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent (Reference 3). The report contained a proposed outline for individual license amendment submittals and discussed the scope and depth of reviews needed and the methodologies used in these reviews. In a letter of September 30, 1991, the NRC approved the program proposed in the report, on the condition that individual power uprate amendment requests meet certain requirements in the document (Reference 2).

The generic BWR power uprate program gives each licensee a consistent means to recover additional generating capacity beyond its current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level for most licensees was based on the vendor-guaranteed power level for the reactor. The difference between the guaranteed power level and the design power level is often referred to as *stretch power*. The design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCS). Therefore, increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment and does not significantly affect the reliability of this equipment.

The licensee's amendment request to increase the current licensed power level of 3323 MWt to a new limit of 3467 MWt represents an approximate 4.3 percent increase in thermal power with a corresponding 5-percent increase in rated steam flow (an increase in vessel steam flow from 14.3 to 15 Mlb/h). NMPC will increase power to the higher level by: (1) increasing the core thermal power to increase steam flow, (2) increasing the feedwater system flow by a corresponding amount, (3) increasing reactor pressure to ensure adequate turbine control margin, (4) not increasing the current maximum core flow, and (5) operating the reactor primarily along extensions of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in Reference 3. The operating pressure will be increased approximately 15 psi to ensure satisfactory pressure control and pressure drop characteristics for the increased steam flow. The increased core power will be achieved by utilizing a flatter radial power distribution while still maintaining limiting fuel bundles within their constraints.

3.0 EVALUATION

The NRC staff reviewed NMPC's request for a NMP-2 power uprate amendment using applicable rules, regulatory guides, sections of the Standard Review Plan (NUREG-0800), and NRC staff positions. The NRC staff also evaluated NMPC's submittal (Reference 1) for compliance with the generic BWR power uprate program as defined in Reference 3. Detailed discussions of individual review topics follow.

3.1 Thermal Limits Assessment

The operating limit minimum critical power ratio (MCPR) is determined on a cycle specific basis from the results of reload analysis, as described in General Electric Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactors Power Uprate," July 1991; and Supplements 1 and 2 (Reference 4). The maximum average planar linear heat generation rate (MAPLHGR) and linear heat generation rate (LHGR) limits will also be maintained as described in this reference. The plant-specific safety evaluation for NMP-2 is contained in References 5 and 6.

3.2 Reactivity Characteristics

3.2.1 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Extended Load Line Limit Analysis (ELLLA). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100% rated power. The changes to the power/flow operating map are consistent with the previously NRC approved generic descriptions given in NEDO-31984.

3.2.2 Stability

Ongoing activities by the BWR Owners' Group and the NRC staff are addressing ways to minimize the occurrence and potential effects of power oscillations that have been observed for certain BWR operating conditions (as required by General Design Criteria 12 of 10 CFR Part 50 Appendix A). GE has documented information and cautions concerning this possibility in Service Information Letter (SIL) 380 and related communications. The NRC has documented its concerns in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin. While a more permanent resolution is being developed, TSs and associated implementing procedures, as requested by the NRC Bulletin, shall be incorporated by NMPC which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Specific operator actions shall be established to provide clear instructions for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions.

The restrictions recommended by NRC Bulletin 88-07 and Supplement 1 to that bulletin will continue to be followed by NMPC for uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWR Owners' Group and the NRC. This is acceptable to the NRC staff.

3.2.3 Control Rod Drives and CRD Hydraulic System

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated steam flow and dome pressure.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion will be slower due to the high pressure. As the scram continues, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Therefore, an increase in the reactor pressure has little effect on scram time. NMPC has indicated that CRD performance during power uprate will meet current TS requirements. NMPC will continue to monitor by various surveillance requirements the scram time performance as required in the plant TSs to ensure that the original licensing basis for the scram system is preserved. For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flow required for CRD cooling and driving are assured by the automatic opening of the system flow control valve, thus compensating for the small increase in reactor pressure. Prior to implementation of power uprate, the flow control valves and CRD pumps will be tested to ensure they are capable of operating within their acceptable range with power uprate. The CRD system should therefore continue to perform all its safety-related functions at uprated power with ICF, and should function adequately during insert and withdraw modes.

3.3 Reactor Coolant System and Connected Systems

3.3.1 Nuclear System Pressure Relief

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) provide this protection. The setpoints for the relief function of the SRVs are increased 15 psi for power uprate.

The operating steam dome pressure is defined to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to uprated power. The uprate dome pressure increase will require a change in the SRV setpoints. The appropriate increase in the SRV setpoints also ensures that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in the number of unnecessary SRV actuations.

3.3.2 Code Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The American Society of Mechanical Engineers (ASME) code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steamline isolation valve (MSIV) closure with a failure of the valve position scram. This transient was analyzed by NMPC with the following assumptions: (1) core power is 3536 MWt (102% of the uprated power of 3467 MWt), (2) end-of-cycle nuclear parameters, (3) two SRVs out-of-service, (4) no credit for the relief mode of the SRVs, (5) TS scram speed, (6) three second MSIV closure time, and (7) initial reactor dome pressure of 1020 psia. The SRV opening pressures were +3% above the nominal setpoint for the available valves. The analysis also assumed credit for the high pressure recirculation pump trip (RPT).

The calculated peak pressure was 1291 psig which is below the ASME allowable of 1375 psig which is acceptable. The number of SRVs which will be assumed to be out-of-service is based on the maximum allowed by TSs. Uprated conditions will produce a higher peak RPV pressure, and with reduced valve grouping, the reload analysis must show that it remains below the 1375 psig ASME code limit. NMPC's analysis plan is acceptable to the NRC staff.

3.3.3 Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with allowance for increased core flow. The cycle-specific core reload analyses will consider the full core flow range, up to 115 Mlb/h. The evaluation by NMPC of the reactor recirculation system performance at uprated power with ICF determined that the core flow can be maintained. The system design pressures for the Reactor Recirculation Control (RRC) System components includes the suction, discharge and flow control valves, recirculation pumps, and piping were evaluated. Raising the steam pressure by 15 psig as a result of power uprate will raise the pump suction pressure by 17 psig and the pump discharge pressure by 45 psig. NMPC states that these increases in normal operating pressures are bounded by the system design pressure. Operation at uprated conditions will increase the RRC pump suction temperature by approximately one degree Fahrenheit which is also bounded by the system design temperature.

The pump speed and flow control valve position runback functions affected by power uprate and ELLL will be changed. The cavitation interlock setpoint will remain the same. NMPC concluded that the changes due to power uprate and ELLL are small and are bounded by the RRC design basis. NMPC should perform power uprate startup testing on the RRC system to demonstrate flow control over the entire pump speed range to enable a complete calibration of the flow control instrumentation including signals to the Process Computer. As stated in NEDO-31897, these tests should also assure no undue vibration occurs at uprate or ELLL conditions. In a letter dated October 6, 1994 (Reference 7), NMPC committed to perform more frequent monitoring of vibrations during the initial power ascension for the uprated power conditions such that vibration levels will be recorded and evaluated prior to and during operation at uprate conditions. This commitment is acceptable to the NRC staff.

3.3.4 Main Steam Isolation Valves (MSIVs)

The MSIVs have been evaluated by NMPC, and are consistent with the bases and conclusions of the generic evaluation. Increased core flow alone does not change the conditions within the main steam lines, and thus cannot affect the MSIVs. Performance will be monitored by surveillance requirements in the TSs to ensure original licensing basis for MSIV's are preserved. This is consistent with the generic evaluation in NEDO-31894, and is acceptable to the NRC staff.

3.3.5 Reactor Core Isolation Cooling System (RCIC)

The RCIC provides core cooling when the RPV is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system. The RCIC system has been evaluated by NMPC, and is consistent with the bases and conclusions of the generic evaluation. The recommendations of GE SIL 377 have been implemented at NMP-2 and NMPC shall complete the additional testing to address all aspects of GE SIL 377. These tests will be conducted during power ascension testing for power uprate. The results of these tests shall be reported in the Startup Test Report required by TS 6.9.1.1. This is acceptable to the NRC staff. The NRC staff requires that NMPC provide assurance that the RCIC system will be capable of injecting its design flow rates at the conditions associated with power uprate. Additionally, NMPC must also provide assurance that the reliability of this system will not be decreased by the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads. NMPC's assurance of RCIC system capability and assurance of its reliability may be provided in the Startup Test Report required by TS 6.9.1.1.

3.3.6 Residual Heat Removal (RHR) System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions.

The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on these operating modes are discussed in the following paragraphs (the LPCI mode is discussed in Section 3.4.3).

3.3.6.1 Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. This increased time is judged to be insignificant.

Regulatory Guide (RG) 1.139, "Guidance for Residual Heat Removal," states that cold shutdown capability (200 °F reactor fluid temperature) should be accomplished within 36 hours. For power uprate, licensee analysis of the alternate path for shutdown cooling based on the criteria of RG 1.139 shows that the reactor can be cooled to 200 °F in less than the 36-hour criterion.

3.3.6.2 Suppression Pool Cooling Mode

The functional design basis for suppression pool cooling mode (SPCM) stated in the Final Safety Analysis Report (FSAR) is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate, since the peak suppression pool temperature analysis by NMPC confirms that the pool temperature will stay below its design limit at uprated conditions.

3.3.6.3 Containment Spray Cooling Mode

The containment spray cooling mode provides water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during post-accident conditions. Power uprate increases the containment spray temperature by only a few degrees. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure since these parameters reach peak values prior to actuation of the containment spray.

3.3.7 Reactor Water Cleanup (RWCU) System

The RWCU system pressure and temperature will increase slightly as a result of power uprate. NMPC has evaluated the impact of these increases and has concluded that uprate will not adversely affect RWCU system integrity. The cleanup effectiveness of the RWCU system may be slightly diminished as a result of increased feedwater flow to the reactor; however, the current limits for reactor water chemistry will remain unchanged with power uprate.

3.4 Engineered Safety Features

3.4.1 Emergency Core Cooling Systems

The effect of power uprate and the increase in RPV dome pressure on each ECCS system is addressed below. Also as discussed in the FSAR, compliance with the net positive suction head (NPSH) requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig and the maximum expected temperature of pumped fluids. The pumps are assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit (212 degrees Fahrenheit). Assuming a loss-of-coolant accident (LOCA) occurs during operation at the uprated power, the suppression pool temperature (208 °F) will remain below its NPSH limit. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

3.4.2 High Pressure Core Spray (HPCS)

The HPCS system was evaluated by NMPC and is consistent with the bases and conclusions contained in the generic evaluation for power uprate. This is acceptable to the NRC staff.

3.4.3 Low Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low pressure portions of the RHR are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. This is consistent with the bases and conclusions of the generic power uprate evaluation.

3.4.4 Low Pressure Core Spray (LPCS) System

The hardware for the low pressure core spray is not affected by power uprate. The upper limit of the low pressure core spray injection setpoints will not be changed for power uprate; therefore, the low pressure portions of this system will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Therefore, since this system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. Also, the impact of power uprate on the long term response to a LOCA will continue to be bounded by the short-term response. The LPCS is bounded by the generic evaluation.

3.4.5 Automatic Depressurization System (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCS failure. This function allows LPCI and core spray (CS) to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for uprate. Plant design requires a minimum flow capacity equivalent to 1 of the 7 SRVs/ADS valves being out-of-service as shown in NMPC analysis for SRV setpoint tolerance and out-of-service analysis to be discussed later in this evaluation. ADS initiates on Low Water Level 1 and a signal that at least one LPCI or LPCS pump is running with permissive from Low Water Level 3. ADS is activated following a maximum time delay of 120 seconds, after the initiating signals if these conditions are met. The ability to perform these functions is not affected by power uprate.

3.4.6 ECCS Performance Evaluation

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50 Appendix K. The General Electric fuel, used in NMP-2 was analyzed by NMPC (Reference 6) with the NRC-approved methods. The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

NMPC used the NRC staff approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-LOCA analysis for NMP-2 was performed by NMPC with GE fuel in accordance with NRC requirements in NEDC-32115P and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K PCT as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases.

The NMP-2 specific analysis was performed at uprated power and the bounding ELLL region using a conservatively high Peak Linear Heat Generation Rate (PLHGR) and a conservatively low MCPR. In addition, some of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The nominal (expected) PCT is 853 °F. The statistical Upper

Bound PCT is below 1240 °F. The Licensing Basis PCT for NMP-2 is 1255 °F which is well below the acceptance criteria of the 10 CFR 50.46 PCT limit of 2200 °F. The analysis also meets the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in Table 6-1 of the NMP-2 Licensing Topical Report. Therefore, NMP-2 meets the NRC S/G-LOCA licensing analysis requirements.

NMPC also reevaluated the ECCS performance for single loop operation (SLO) using the S/G - LOCA methodology. The design-basis accident (DBA) size break is also limiting for SLO. Using the same assumptions in the S/G - LOCA calculation with no MAPLHGR reduction, yields a calculated nominal PCT of 1100 °F and 1417 °F, depending on the type of fuel. Since the PCT was below the 10 CFR 50.46 limit of 2200 °F, NMPC claimed that no MAPLHGR reduction is required for SLO. The NRC staff asked NMPC to reconcile the fact that the S/G - LOCA analysis PCT results for SLO were higher than those presented for two loop operation, and no statistical analysis of the Upper Bound PCT had been provided for this case. NMPC reviewed this NRC staff question, and has stated in Reference 7 that the SLO PCT for NMP-2 are above the two-loop PCTs because no SLO APLHGR restrictions were applied, full power was assumed, and immediate dryout was assumed. The current NMP-2 T/S applies a multiplier to the APLHGR for SLO. NMPC has taken the approach of applying applicable SLO APLHGR multipliers for each fuel type which will be presented in the Core Operating Limits Report (COLR). The SLO PCTs are lower than the two loop PCTs when these multipliers are applied. This is acceptable to the NRC staff.

The impact of Increased Core Flow (ICF), up to 115 Mlb/h, on LOCA results was evaluated at the 3629 MWt power level using S/G-LOCA methodology for NMP-2. For a DBA recirculation line break with the same single failure (HPCS diesel) and using the same Appendix K and nominal assumptions the results show a decrease in the nominal PCT when compared to the base case.

This decrease in PCT for the nominal ICF case is due to: (1) the better heat transfer during flow coast-down from the higher initial flow; and (2) less subcooling in the downcomer which results in reduced break flow and later core uncovery.

3.5 Reactor Safety Performance Features

3.5.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. NMPC will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients. The limiting events for NMP-2 were identified. These are the same as those in the generic report on power uprate. The generic guidelines also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. Representative changes in core CPR's for the normally analyzed transients were provided; however, specific core operating limits will be supplied for each specific fuel cycle. The power uprate with ELLL operation were presented for a representative core using the GEMINI transient analysis methods listed in the generic report.

The Safety Limit Minimum Critical Power Ratio (SLMCPR) will be confirmed for each operating fuel cycle, at the time of the reload analysis, using the NRC-approved SNP methodology. The SLMCPR used in the analysis to calculate the operating limit MCPR was 1.07.

The limiting transients for each category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The results from these analyses developed the licensing basis for transient analyses at uprated power with ELLL operation. The limiting transient results were presented in NMPC submittal in Table 9-2. These were the applicable transients as specified in the generic power uprate guidelines report (NEDC-31897). Cycle specific analyses will be done at each reload and will be a part of the COLR developed by NMPC.

This is acceptable to the NRC staff and will be reviewed as part of NMPC's reload submittal.

3.5.2 Anticipated Transients Without Scram (ATWS)

A generic evaluation for the ATWS events is presented in Section 3.7 of Supplement 2 of the Generic Report (NEDC-31984) for BWR/5 reactors. This evaluation concludes that the results of an ATWS event are acceptable for the fuel, RPV, and the containment response for a power uprate of 4.3%. The NMP-2 power increase is 4.3%, which is within the generic evaluation. Therefore, the ATWS analysis is acceptable for NMP-2.

3.5.3 Station Blackout (SBO)

The NMP-2 SBO plant responses were evaluated at a steam flow increase of 105% for power uprate. This corresponds to an increase of reactor thermal power of 3536 MWt. The NMP-2 response to a postulated SBO uses the RCIC and HPCS for core cooling. A coping evaluation was performed to demonstrate performance, based on HPCS with backup provided by the RCIC system. The coping time remains unchanged for power uprate. However, the RCIC system is the preferred source for initial operation. No changes to the systems or equipment used to respond to a SBO are necessary due to power uprate. The analysis was done at uprate and ELLL operating conditions. The suppression pool temperature remained within design conditions, therefore all equipment that takes suction from the suppression pool will continue to operate when power is restored.

The evaluation assumes a reactor power of 3536 MWt at an operating pressure of 1035 psia. The individual considerations evaluated for power uprate included the following: the regulatory basis; the event scenario; condensate inventory and reactor coolant inventory; station battery load; compressed air supply; and loss of ventilation to the control room, reactor protection system rooms and switchgear rooms, HPCS pump and auxiliary rooms, RCIC room, containment, suppression pool and spent fuel pool. The SBO analysis is acceptable to the NRC staff.

3.6 Containment Evaluation

The NMP-2 updated safety analysis report (USAR) provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with power uprate changes some of the conditions for the containment analyses. Section 5.10.2 of Topical Report NEDC-31897, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate" requires the power uprate applicant to show acceptability of the uprated power level for: (1) containment pressures and temperatures, (2) LOCA containment dynamic loads, and (3) safety-relief valve dynamic loads. Appendix G of NEDC-31897 prescribes the approach to be used by power uprate applicants for performing required plant-specific analyses. NMPC did the necessary analyses and discussed the results in the application.

Appendix G of NEDC-31897 states that the applicant will analyze short-term containment responses using the staff-approved M3CPT code. M3CPT is used to analyze the period from when the break begins to when pool cooling begins. M3CPT generates data on the response of containment pressure and temperature (Section 3.6.1), dynamic loads analyses (Section 3.6.2) and for equipment qualification analyses (Section 3.13).

Appendix G of NEDC-31897 also states that the applicant will perform long-term containment heatup (suppression pool temperature) analyses for the limiting safety analysis report events to show that the pool temperatures will remain within limits for:

Containment design temperature,
local pool temperature,
Net positive suction head (NPSH),
pump seals, piping design temperature, and other limits

These analyses will use the SHEX code and ANS 5.1-1979 decay heat assumptions consistent with the NRC staff's letter to Gary L. Sozzi of July 13, 1993. SHEX, which is partially based on M3CPT, is a long term code to analyze the period from when the break begins until after peak pool heatup.

3.6.1 Containment Pressure and Temperature Response

Short-term and long-term containment analyses of containment pressure and temperature response following a large break inside the drywell are documented in the USAR. The short-term analysis is performed primarily to determine the peak drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment after a design basis accident (DBA) LOCA. The long-term analysis is performed primarily to determine the peak pool temperature response.

3.6.1.1 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

NMPC indicated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA at 102 percent of the uprated power using the SHEX code and ANS 5.1 decay heat assumptions prescribed by NEDC-31897. The analyses have been performed using the more realistic RHR pool cooling capability than that which was used in the original analyses (K-factor=240.2 Btu/sec-° F vs. 199.2 Btu/sec-° F), but also with a higher service water temperature (82 °F vs 77 °F). The NRC staff has approved the use of the higher K factor and service water temperature in a safety evaluation to Amendment No. 3 dated April 11, 1988. All other key input parameters for power uprate analyses were essentially the same as those for the original analyses. For the power uprate, the DBA-LOCA peak suppression pool temperature was calculated to be 207.9 °F. This temperature is approximately 1 °F higher than the value given in the USAR but is within suppression pool design temperature limit of 212 °F and meets the ECCS pumps NPSH requirements.

NMPC indicated that the long-term bulk pool temperature response was also evaluated for the non-LOCA limiting event which assumes reactor isolation with only one RHR heat exchanger available to accommodate SRV discharge to the suppression pool. The peak bulk suppression pool temperature calculated with 102 percent of the uprated power was 210.9 °F. This temperature is approximately 2 °F higher than the value obtained with the current power but is within the suppression pool design value of 212 °F.

Based on the results of these analyses, the NRC staff concludes that the bulk suppression pool temperature response remains acceptable after power uprate.

(2) Local Pool Temperature with SRV Discharge

A local pool temperature limit for SRV discharge is specified in NUREG-0783 because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. NMPC indicated that since the NMP-2 has quenchers, no evaluation of this limit is considered necessary. Elimination of this limit for plants with quenchers on the SRV discharge lines is justified in GE Report NEDO-30832, "Elimination of Limits on Local Suppression Pool Temperature for SRV Discharge with quenchers." NEDO-30832 has been evaluated and approved by the NRC staff (SE dated August 29, 1994). However, the local pool temperature has been evaluated at uprated power, and was found to be acceptable with respect to NUREG-0783 limit.

Based on the above, the NRC staff concludes that the local pool temperature limit will remain acceptable after power uprate.

3.6.1.2 Containment Gas Temperature Response

NMPC indicated that the containment drywell design temperature of 340 °F was determined based on a bounding analysis of the superheated gas temperature which can be reached with blowdown of steam to the drywell during a LOCA. The expansion of the reactor steam under these conditions will result in a calculated peak drywell temperature of 325.8 °F. Assuming that there is a 6-hour cooldown period required to completely depressurize the reactor vessel based on a controlled 100 °F/hr cooldown rate, the small steamline break analysis shows the peak value to be approximately 270 °F at current power. Small steamline breaks in the drywell impose the most severe drywell temperature conditions. The changes in the reactor vessel conditions with power uprate will increase the calculated long-term peak drywell gas temperature response during a small-break LOCA by a maximum of a few degrees but will not exceed the drywell design value of 340 °F. Therefore, the drywell gas temperature response after power uprate will remain below the containment design temperature of 340 °F.

NMPC indicated that the wetwell gas space peak temperature response was calculated assuming thermal equilibrium between the pool and wetwell gas space. The reanalysis has shown that the maximum bulk pool temperature will reach 207.9 °F after LOCA and 210.9 °F after alternate shutdown due to power uprate. Therefore, the maximum wetwell gas space temperature due to power uprate will remain below the wetwell design temperature of 270 °F.

Based on its review, the NRC staff concludes that the containment drywell and wetwell gas temperature response will remain acceptable after power uprate.

3.6.1.3 Short Term Containment Pressure Response

NMPC indicated that the short-term containment response analyses were performed for the limiting DBA-LOCA, which assumes a double-ended guillotine break of a recirculation suction line to demonstrate that power uprate operation will not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and differential pressure between the drywell and wetwell occur. These analyses were performed at 102% of the uprated power level, using the GE M3CPT computer code. The reanalysis predicted a maximum containment pressure of 36.8 psig which remains below the containment design pressure of 45 psig. The reanalysis also predicted a maximum drywell-to-wetwell pressure difference of 16.3 psid which remains below the design limit of 25 psid.

TSs definitions, limiting conditions for operation, surveillance requirements and bases relating to the current 39.75 psig value of P_a will not be revised as it remains higher than the maximum containment pressure of 36.8 psig calculated for the power uprate.

Based on its review, the NRC staff concludes that the containment pressure response following a postulated LOCA will remain acceptable after power uprate.

3.6.1.4 Steam Bypass Case

NMPC indicated that the steam bypass of the suppression pool due to a leakage between the drywell and the wetwell airspace during a LOCA event was analyzed to ensure that the primary containment design pressure of 45 psig is not exceeded. The amount of steam bypass leakage is determined by the magnitude and duration of the pressure difference between the drywell and the wetwell during a LOCA (governed by the vent submergence), and by the leakage flow area. These parameters are not affected by reactor power. The assumed time of 30 minutes required for the operator to initiate containment spray operation is not changed. Power uprate will only influence the suppression pool temperature, and subsequently, the primary containment pressure. A bounding evaluation estimated an increase of approximately 0.2 psi in the peak drywell pressure based on the increase in the bulk suppression pool temperature prior to initiation of containment sprays at 30 minutes. The 0.2 psi increase in the peak primary containment pressure due to power uprate will not result in a peak primary containment pressure which exceeds the design value of 45 psig. Assuming the 0.2 psi increase in the peak drywell pressure, the maximum allowable ($A/K^{0.5}$) steam bypass capacity is reduced from 0.057 sq. ft. to about 0.056 sq. ft. (USAR Figure 6.2-28) but remains above the 0.054 sq. ft. value used as the basis for the current TS for allowable bypass leakage. The evaluation shows that the power uprate has negligible impact on the suppression pool steam bypass effects.

Based on the above, the NRC staff concludes that the steam bypass response will remain acceptable after power uprate.

3.6.2 Containment Dynamic Loads

3.6.2.1 LOCA Containment Dynamic Loads

NEDC-31897 requires that the power uprate applicant determine if the containment pressure, temperature and vent flow conditions, calculated with the M3CPT code for power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads were based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by power uprate and thus do not require further analysis.

NMPC indicated that the LOCA dynamic loads which are considered in the power uprate evaluation include pool swell, condensation oscillation (CO), and chugging. The initial drywell pressurization rate used to define the pool swell load bounds the value calculated with the uprated power. The short-term containment response conditions for vent flow rate and pool temperature with power uprate are within the range of test conditions used to define the CO loads. The containment conditions with power uprate in which chugging would occur are within the range of test conditions used to define the chugging loads. Therefore, the LOCA dynamic loads for NMP-2 are not impacted by power uprate.

Based on the above, the NRC staff concludes that the LOCA containment dynamic loads will remain acceptable after power uprate.

3.6.2.2 Safety Relief Valve (SRV) Containment Dynamic Loads

The SRV containment dynamic loads include discharge line loads, pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by SRV opening setpoints, discharge line configuration and suppression pool configuration. Of these parameters only the SRV setpoint is affected by power uprate. NEDC-31897 states that if the SRV setpoints are increased, the power uprate applicant will attempt to show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

NMPC indicated that the highest SRV opening setpoint with power uprate will be 1241 psig. The SRV setpoint which was the basis for the SRVDL loads and the SRV loads on the suppression pool boundary and submerged structures is 1261 psig. Since the highest setpoint with power uprate remains lower than the setpoint used to define the SRV loads, power uprate does not impact the SRV definitions for the first actuation of SRVs. The water leg prior to SRV opening used to define the subsequent actuation loads conservatively assumed the maximum calculated SRVDL reflood height. This is not impacted by power uprate. Therefore, there will be no effect of power uprate on the water leg prior to SRV opening and no impact of power uprate on the subsequent actuation loads. The SRV containment dynamic loads will remain below their original design values after power uprate.

Based on the above, the NRC staff concludes that the SRV containment dynamic loads will remain acceptable after power uprate.

3.6.2.3 Subcompartment Pressurization

NMPC indicated that the design loads on the annulus between the biological shield wall and vessel and the drywell head due to a postulated pipe break in the annulus were evaluated for the limiting subcompartment pressurization event at uprated conditions. The values used for the power uprate evaluation at 102% of the uprated power are not significantly changed from the values used for original analysis at 104.3% of current power. The subcompartment pressurization loads are not significantly affected by power uprate and remain acceptable. It is also noted that the NEDC-31897 methodology does not require subcompartment reanalysis. Based on the above, the NRC staff concludes that the subcompartment pressurization effects will remain acceptable after power uprate.

3.6.3 Containment Isolation

The NEDC-31897 methodology does not address a need for reanalysis of the isolation system. The system designs for containment isolation are not affected by power uprate. The capability of the actuation devices to perform with uprated pressure and flow will comply for acceptability in response to

Generic Letter 89-10 at uprated conditions. Based on its review, the NRC staff finds that the operation of the plant at uprated power level will not impact the containment isolation system.

3.6.4 Post-LOCA Combustible Gas Control

NMPC indicated that the hydrogen recombiners are provided to maintain the containment atmosphere as a non-combustible mixture after DBA-LOCA. The combustibility of the post-LOCA containment atmosphere is controlled by the concentration of oxygen. As a result of power uprate, the post-LOCA production of oxygen and hydrogen by radiolysis will increase proportionally with power level. The original evaluation of the system was performed at 3467 MWt, the evaluation at uprated operation increases only by 2%. Sufficient capacity exists in the combustible gas control system to accommodate the slightly increased oxygen and hydrogen production. Also, recombiner operation is controlled procedurally based on gas concentration in the containment. Based on its review, the NRC staff concludes that the post-LOCA combustible gas control will remain acceptable after uprated power.

3.7 Standby Gas Treatment System (SGTS)

The SGTS is designed to minimize offsite dose rates during venting and purging of both the primary and secondary containment atmosphere under accident or abnormal conditions, while containing airborne particulate and halogens that might be present. The SGTS consists of two identical, parallel, physically separated, 100-percent capacity air filtration assemblies with associated piping, valves, controls, and centrifugal exhaust fans. Effluents from the SGTS connect to a common exhaust line discharging to the exhaust tunnel leading to the main tack. The SGTS draws air from the reactor building.

Following a postulated accident, the SGTS is started, taking over from the normal ventilation system which has been maintaining secondary containment at a slightly negative pressure, $\leq$ -0.25 inch water gauge (WG). Maintaining this negative pressure serves to prevent unfiltered release of radioactive material from the secondary containment to the environment. During the transfer to SGTS operation, pressure rises momentarily until the SGTS, together with the Category I unit coolers, reestablishes pressure $\leq$ -0.25 inch WG.

NMPC indicated that appropriate differential temperature requirements will be maintained for uprated operation to ensure that the secondary containment atmosphere temperature is sufficiently above the available service water temperature so that the negative pressure is restored within the time period assumed in the radiological evaluations. The air-flow capacity of the SGTS was selected to accommodate the in-leakage equivalent to one secondary containment air volume change per day and thereby maintain the reactor building at the desired negative pressure. The SGTS capability remains adequate for uprated operation in conjunction with appropriate differential temperature requirements.

NMPC also indicated that the charcoal filter beds are not significantly affected by uprated power level operation. The SGTS is designed to be in compliance with RG 1.52 (Rev. 2) with numerous minor exceptions, including charcoal loading capacity. The SGTS is designed for a charcoal loading capacity of 10mgI/gC as compared to a value of 2.5mgI/gC per RG 1.52 (Rev. 2), and meets the design requirements for 30-day and 100-day LOCA scenarios. The total post-LOCA iodine loading increases less than 4.3% at the uprated conditions and remains within the 10mgI/gC loading limit of the system.

The NRC staff reviewed NMPC's use of 10mgI/gC loading capacity. This exception along with numerous other exceptions to RG 1.52 was submitted to the NRC staff in the FSAR prior to issuance of the NMP-2 operating license. The NRC staff's safety evaluation accepted all exceptions to RG 1.52 but did not discuss the basis for acceptance. The only exception of concern to the NRC staff for power uprate was the charcoal loading capacity.

NMPC provided additional justification for the deviation to the charcoal loading capacity recommendation in RG 1.52 in their letter dated September 16, 1994. NMPC states in their letter that adsorbed iodine in the charcoal would not generate heat at a sufficient rate to result in either combustion of the charcoal or temperatures high enough to cause significant desorption of the iodines. The charcoal adsorption capacity of 10 mgI/gC is within the adsorption capacity of the activated carbon used in the SGTS with respect to loading capacity and adsorption efficiency. The carbon capacity is supported

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

April 28, 1995

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M87088)

Dear Mr. Sylvia:

The Commission has issued the enclosed Amendment No. 66 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 July 22, 1993, as supplemented by letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995.

The amendment modifies Facility Operating License No. NPF-69 and the NMP-2 TSs to authorize an increase in the maximum power level of NMP-2 from 3323 megawatts thermal (MWt) to 3467 MWt. The amendment also approves changes to the TSs to implement uprated power operation.

The amendment is effective as of its date of issuance and is to be implemented prior to restart from refueling outage 4.

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

Sincerely,

Frehn 83852-104

Original signed by:
Gordon E. Edison, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures: 1. Amendment No. 66 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

*See previous concurrence

DOCUMENT NAME: NM287088.AMD

OFFICE	LA:PDI-1	PM:PDI-1	BC:EMFB	BC:SRXB *	BC:TERB *
NAME	SLittle	GEdison:cn	~~[illegible]~~ CThomas	RJones	CMiller
DATE	04/13/95	04/13/95	04/13/95	04/14/95	04/17/95
OFFICE	BC:EMEB *	BC:SPLB *	BC:SCSB *	BC:HICB *	OGC
NAME	RWessman	CMcCracken	RBarrett	JWermiel	
DATE	04/ /95	04/ /95	04/12/95	04/13/95	04/ /95
OFFICE	D:PDI-1	D:DRPE	ADP:NRR	D:NRR	
NAME	LMarsh	SVarga	RZimmerman	WRussell	
DATE	04/21/95	04/ /95	04/27/95	04/28/95	4/ /

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(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. 66 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 and is to be implemented prior to startup from refueling outage 4.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments: 1. Pages 3 and 5 of License*
2. Changes to the Technical Specifications

Date of Issuance: April 28, 1995

*Pages 3 and 5 are attached, for convenience, for the composite license to reflect this change.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 66 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 July 22, 1993 (Reference 1), as supplemented by letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995, Niagara Mohawk Power Corporation (the licensee or NMPC) submitted a request for changes to the Nine Mile Point Nuclear Station, Unit 2 (NMP-2), Facility Operating License (NPF-69) and for changes to the NMP-2 Technical Specifications (TSs). The request would increase the licensed thermal power level of the NMP-2 reactor from the current limit of 3323 megawatts thermal (MWt) to 3467 MWt. The request would also approve changes to the TSs to implement uprated power operation. This request is in accordance with the generic boiling-water reactor (BWR) power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter dated September 30, 1991 (Reference 2). NMPC's letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

On December 28, 1990, GE submitted GE Licensing Topical Report (LTR) NEDC-31897P-1, in which it proposed to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent (Reference 3). The report contained a proposed outline for individual license amendment submittals and discussed the scope and depth of reviews needed and the methodologies used in these reviews. In a letter of September 30, 1991, the NRC approved the program proposed in the report, on the condition that individual power uprate amendment requests meet certain requirements in the document (Reference 2).

The generic BWR power uprate program gives each licensee a consistent means to recover additional generating capacity beyond its current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level for most licensees was based on the vendor-guaranteed power level for the reactor. The difference between the guaranteed power level and the design power level is often referred to as *stretch power*. The design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCS). Therefore, increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment and does not significantly affect the reliability of this equipment.

The licensee's amendment request to increase the current licensed power level of 3323 MWt to a new limit of 3467 MWt represents an approximate 4.3 percent increase in thermal power with a corresponding 5-percent increase in rated steam flow (an increase in vessel steam flow from 14.3 to 15 Mlb/h). NMPC will increase power to the higher level by: (1) increasing the core thermal power to increase steam flow, (2) increasing the feedwater system flow by a corresponding amount, (3) increasing reactor pressure to ensure adequate turbine control margin, (4) not increasing the current maximum core flow, and (5) operating the reactor primarily along extensions of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in Reference 3. The operating pressure will be increased approximately 15 psi to ensure satisfactory pressure control and pressure drop characteristics for the increased steam flow. The increased core power will be achieved by utilizing a flatter radial power distribution while still maintaining limiting fuel bundles within their constraints.

3.0 EVALUATION

The NRC staff reviewed NMPC's request for a NMP-2 power uprate amendment using applicable rules, regulatory guides, sections of the Standard Review Plan (NUREG-0800), and NRC staff positions. The NRC staff also evaluated NMPC's submittal (Reference 1) for compliance with the generic BWR power uprate program as defined in Reference 3. Detailed discussions of individual review topics follow.

3.1 Thermal Limits Assessment

The operating limit minimum critical power ratio (MCPR) is determined on a cycle specific basis from the results of reload analysis, as described in General Electric Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactors Power Uprate," July 1991; and Supplements 1 and 2 (Reference 4). The maximum average planar linear heat generation rate (MAPLHGR) and linear heat generation rate (LHGR) limits will also be maintained as described in this reference. The plant-specific safety evaluation for NMP-2 is contained in References 5 and 6.

3.2 Reactivity Characteristics

3.2.1 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Extended Load Line Limit Analysis (ELLLA). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100% rated power. The changes to the power/flow operating map are consistent with the previously NRC approved generic descriptions given in NEDO-31984.

3.2.2 Stability

Ongoing activities by the BWR Owners' Group and the NRC staff are addressing ways to minimize the occurrence and potential effects of power oscillations that have been observed for certain BWR operating conditions (as required by General Design Criteria 12 of 10 CFR Part 50 Appendix A). GE has documented information and cautions concerning this possibility in Service Information Letter (SIL) 380 and related communications. The NRC has documented its concerns in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin. While a more permanent resolution is being developed, TSs and associated implementing procedures, as requested by the NRC Bulletin, shall be incorporated by NMPC which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Specific operator actions shall be established to provide clear instructions for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions.

The restrictions recommended by NRC Bulletin 88-07 and Supplement 1 to that bulletin will continue to be followed by NMPC for uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWR Owners' Group and the NRC. This is acceptable to the NRC staff.

3.2.3 Control Rod Drives and CRD Hydraulic System

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated steam flow and dome pressure.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion will be slower due to the high pressure. As the scram continues, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Therefore, an increase in the reactor pressure has little effect on scram time. NMPC has indicated that CRD performance during power uprate will meet current TS requirements. NMPC will continue to monitor by various surveillance requirements the scram time performance as required in the plant TSs to ensure that the original licensing basis for the scram system is preserved. For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flow required for CRD cooling and driving are assured by the automatic opening of the system flow control valve, thus compensating for the small increase in reactor pressure. Prior to implementation of power uprate, the flow control valves and CRD pumps will be tested to ensure they are capable of operating within their acceptable range with power uprate. The CRD system should therefore continue to perform all its safety-related functions at uprated power with ICF, and should function adequately during insert and withdraw modes.

3.3 Reactor Coolant System and Connected Systems

3.3.1 Nuclear System Pressure Relief

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) provide this protection. The setpoints for the relief function of the SRVs are increased 15 psi for power uprate.

The operating steam dome pressure is defined to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to uprated power. The uprate dome pressure increase will require a change in the SRV setpoints. The appropriate increase in the SRV setpoints also ensures that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in the number of unnecessary SRV actuations.

3.3.2 Code Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The American Society of Mechanical Engineers (ASME) code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steamline isolation valve (MSIV) closure with a failure of the valve position scram. This transient was analyzed by NMPC with the following assumptions: (1) core power is 3536 MWt (102% of the uprated power of 3467 MWt), (2) end-of-cycle nuclear parameters, (3) two SRVs out-of-service, (4) no credit for the relief mode of the SRVs, (5) TS scram speed, (6) three second MSIV closure time, and (7) initial reactor dome pressure of 1020 psia. The SRV opening pressures were +3% above the nominal setpoint for the available valves. The analysis also assumed credit for the high pressure recirculation pump trip (RPT).

The calculated peak pressure was 1291 psig which is below the ASME allowable of 1375 psig which is acceptable. The number of SRVs which will be assumed to be out-of-service is based on the maximum allowed by TSs. Uprated conditions will produce a higher peak RPV pressure, and with reduced valve grouping, the reload analysis must show that it remains below the 1375 psig ASME code limit. NMPC's analysis plan is acceptable to the NRC staff.

3.3.3 Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with allowance for increased core flow. The cycle-specific core reload analyses will consider the full core flow range, up to 115 Mlb/h. The evaluation by NMPC of the reactor recirculation system performance at uprated power with ICF determined that the core flow can be maintained. The system design pressures for the Reactor Recirculation Control (RRC) System components includes the suction, discharge and flow control valves, recirculation pumps, and piping were evaluated. Raising the steam pressure by 15 psig as a result of power uprate will raise the pump suction pressure by 17 psig and the pump discharge pressure by 45 psig. NMPC states that these increases in normal operating pressures are bounded by the system design pressure. Operation at uprated conditions will increase the RRC pump suction temperature by approximately one degree Fahrenheit which is also bounded by the system design temperature.

The pump speed and flow control valve position runback functions affected by power uprate and ELLL will be changed. The cavitation interlock setpoint will remain the same. NMPC concluded that the changes due to power uprate and ELLL are small and are bounded by the RRC design basis. NMPC should perform power uprate startup testing on the RRC system to demonstrate flow control over the entire pump speed range to enable a complete calibration of the flow control instrumentation including signals to the Process Computer. As stated in NEDO-31897, these tests should also assure no undue vibration occurs at uprate or ELLL conditions. In a letter dated October 6, 1994 (Reference 7), NMPC committed to perform more frequent monitoring of vibrations during the initial power ascension for the uprated power conditions such that vibration levels will be recorded and evaluated prior to and during operation at uprate conditions. This commitment is acceptable to the NRC staff.

3.3.4 Main Steam Isolation Valves (MSIVs)

The MSIVs have been evaluated by NMPC, and are consistent with the bases and conclusions of the generic evaluation. Increased core flow alone does not change the conditions within the main steam lines, and thus cannot affect the MSIVs. Performance will be monitored by surveillance requirements in the TSs to ensure original licensing basis for MSIV's are preserved. This is consistent with the generic evaluation in NEDO-31894, and is acceptable to the NRC staff.

3.3.5 Reactor Core Isolation Cooling System (RCIC)

The RCIC provides core cooling when the RPV is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system. The RCIC system has been evaluated by NMPC, and is consistent with the bases and conclusions of the generic evaluation. The recommendations of GE SIL 377 have been implemented at NMP-2 and NMPC shall complete the additional testing to address all aspects of GE SIL 377. These tests will be conducted during power ascension testing for power uprate. The results of these tests shall be reported in the Startup Test Report required by TS 6.9.1.1. This is acceptable to the NRC staff. The NRC staff requires that NMPC provide assurance that the RCIC system will be capable of injecting its design flow rates at the conditions associated with power uprate. Additionally, NMPC must also provide assurance that the reliability of this system will not be decreased by the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads. NMPC's assurance of RCIC system capability and assurance of its reliability may be provided in the Startup Test Report required by TS 6.9.1.1.

3.3.6 Residual Heat Removal (RHR) System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions.

The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on these operating modes are discussed in the following paragraphs (the LPCI mode is discussed in Section 3.4.3).

3.3.6.1 Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. This increased time is judged to be insignificant.

Regulatory Guide (RG) 1.139, "Guidance for Residual Heat Removal," states that cold shutdown capability (200 °F reactor fluid temperature) should be accomplished within 36 hours. For power uprate, licensee analysis of the alternate path for shutdown cooling based on the criteria of RG 1.139 shows that the reactor can be cooled to 200 °F in less than the 36-hour criterion.

3.3.6.2 Suppression Pool Cooling Mode

The functional design basis for suppression pool cooling mode (SPCM) stated in the Final Safety Analysis Report (FSAR) is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate, since the peak suppression pool temperature analysis by NMPC confirms that the pool temperature will stay below its design limit at uprated conditions.

3.3.6.3 Containment Spray Cooling Mode

The containment spray cooling mode provides water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during post-accident conditions. Power uprate increases the containment spray temperature by only a few degrees. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure since these parameters reach peak values prior to actuation of the containment spray.

3.3.7 Reactor Water Cleanup (RWCU) System

The RWCU system pressure and temperature will increase slightly as a result of power uprate. NMPC has evaluated the impact of these increases and has concluded that uprate will not adversely affect RWCU system integrity. The cleanup effectiveness of the RWCU system may be slightly diminished as a result of increased feedwater flow to the reactor; however, the current limits for reactor water chemistry will remain unchanged with power uprate.

3.4 Engineered Safety Features

3.4.1 Emergency Core Cooling Systems

The effect of power uprate and the increase in RPV dome pressure on each ECCS system is addressed below. Also as discussed in the FSAR, compliance with the net positive suction head (NPSH) requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig and the maximum expected temperature of pumped fluids. The pumps are assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit (212 degrees Fahrenheit). Assuming a loss-of-coolant accident (LOCA) occurs during operation at the uprated power, the suppression pool temperature (208 °F) will remain below its NPSH limit. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

3.4.2 High Pressure Core Spray (HPCS)

The HPCS system was evaluated by NMPC and is consistent with the bases and conclusions contained in the generic evaluation for power uprate. This is acceptable to the NRC staff.

3.4.3 Low Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low pressure portions of the RHR are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. This is consistent with the bases and conclusions of the generic power uprate evaluation.

3.4.4 Low Pressure Core Spray (LPCS) System

The hardware for the low pressure core spray is not affected by power uprate. The upper limit of the low pressure core spray injection setpoints will not be changed for power uprate; therefore, the low pressure portions of this system will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Therefore, since this system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. Also, the impact of power uprate on the long term response to a LOCA will continue to be bounded by the short-term response. The LPCS is bounded by the generic evaluation.

3.4.5 Automatic Depressurization System (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCS failure. This function allows LPCI and core spray (CS) to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for uprate. Plant design requires a minimum flow capacity equivalent to 1 of the 7 SRVs/ADS valves being out-of-service as shown in NMPC analysis for SRV setpoint tolerance and out-of-service analysis to be discussed later in this evaluation. ADS initiates on Low Water Level 1 and a signal that at least one LPCI or LPCS pump is running with permissive from Low Water Level 3. ADS is activated following a maximum time delay of 120 seconds, after the initiating signals if these conditions are met. The ability to perform these functions is not affected by power uprate.

3.4.6 ECCS Performance Evaluation

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50 Appendix K. The General Electric fuel, used in NMP-2 was analyzed by NMPC (Reference 6) with the NRC-approved methods. The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

NMPC used the NRC staff approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-LOCA analysis for NMP-2 was performed by NMPC with GE fuel in accordance with NRC requirements in NEDC-32115P and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K PCT as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases.

The NMP-2 specific analysis was performed at uprated power and the bounding ELLL region using a conservatively high Peak Linear Heat Generation Rate (PLHGR) and a conservatively low MCPR. In addition, some of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The nominal (expected) PCT is 853 °F. The statistical Upper

Bound PCT is below 1240 °F. The Licensing Basis PCT for NMP-2 is 1255 °F which is well below the acceptance criteria of the 10 CFR 50.46 PCT limit of 2200 °F. The analysis also meets the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in Table 6-1 of the NMP-2 Licensing Topical Report. Therefore, NMP-2 meets the NRC S/G-LOCA licensing analysis requirements.

NMPC also reevaluated the ECCS performance for single loop operation (SLO) using the S/G - LOCA methodology. The design-basis accident (DBA) size break is also limiting for SLO. Using the same assumptions in the S/G - LOCA calculation with no MAPLHGR reduction, yields a calculated nominal PCT of 1100 °F and 1417 °F, depending on the type of fuel. Since the PCT was below the 10 CFR 50.46 limit of 2200 °F, NMPC claimed that no MAPLHGR reduction is required for SLO. The NRC staff asked NMPC to reconcile the fact that the S/G - LOCA analysis PCT results for SLO were higher than those presented for two loop operation, and no statistical analysis of the Upper Bound PCT had been provided for this case. NMPC reviewed this NRC staff question, and has stated in Reference 7 that the SLO PCT for NMP-2 are above the two-loop PCTs because no SLO APLHGR restrictions were applied, full power was assumed, and immediate dryout was assumed. The current NMP-2 T/S applies a multiplier to the APLHGR for SLO. NMPC has taken the approach of applying applicable SLO APLHGR multipliers for each fuel type which will be presented in the Core Operating Limits Report (COLR). The SLO PCTs are lower than the two loop PCTs when these multipliers are applied. This is acceptable to the NRC staff.

The impact of Increased Core Flow (ICF), up to 115 Mlb/h, on LOCA results was evaluated at the 3629 MWt power level using S/G-LOCA methodology for NMP-2. For a DBA recirculation line break with the same single failure (HPCS diesel) and using the same Appendix K and nominal assumptions the results show a decrease in the nominal PCT when compared to the base case.

This decrease in PCT for the nominal ICF case is due to: (1) the better heat transfer during flow coast-down from the higher initial flow; and (2) less subcooling in the downcomer which results in reduced break flow and later core uncovery.

3.5 Reactor Safety Performance Features

3.5.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. NMPC will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients. The limiting events for NMP-2 were identified. These are the same as those in the generic report on power uprate. The generic guidelines also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. Representative changes in core CPR's for the normally analyzed transients were provided; however, specific core operating limits will be supplied for each specific fuel cycle. The power uprate with ELLL operation were presented for a representative core using the GEMINI transient analysis methods listed in the generic report.

The Safety Limit Minimum Critical Power Ratio (SLMCPR) will be confirmed for each operating fuel cycle, at the time of the reload analysis, using the NRC-approved SNP methodology. The SLMCPR used in the analysis to calculate the operating limit MCPR was 1.07.

The limiting transients for each category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The results from these analyses developed the licensing basis for transient analyses at uprated power with ELLL operation. The limiting transient results were presented in NMPC submittal in Table 9-2. These were the applicable transients as specified in the generic power uprate guidelines report (NEDC-31897). Cycle specific analyses will be done at each reload and will be a part of the COLR developed by NMPC.

This is acceptable to the NRC staff and will be reviewed as part of NMPC's reload submittal.

3.5.2 Anticipated Transients Without Scram (ATWS)

A generic evaluation for the ATWS events is presented in Section 3.7 of Supplement 2 of the Generic Report (NEDC-31984) for BWR/5 reactors. This evaluation concludes that the results of an ATWS event are acceptable for the fuel, RPV, and the containment response for a power uprate of 4.3%. The NMP-2 power increase is 4.3%, which is within the generic evaluation. Therefore, the ATWS analysis is acceptable for NMP-2.

3.5.3 Station Blackout (SBO)

The NMP-2 SBO plant responses were evaluated at a steam flow increase of 105% for power uprate. This corresponds to an increase of reactor thermal power of 3536 MWt. The NMP-2 response to a postulated SBO uses the RCIC and HPCS for core cooling. A coping evaluation was performed to demonstrate performance, based on HPCS with backup provided by the RCIC system. The coping time remains unchanged for power uprate. However, the RCIC system is the preferred source for initial operation. No changes to the systems or equipment used to respond to a SBO are necessary due to power uprate. The analysis was done at uprate and ELLL operating conditions. The suppression pool temperature remained within design conditions, therefore all equipment that takes suction from the suppression pool will continue to operate when power is restored.

The evaluation assumes a reactor power of 3536 MWt at an operating pressure of 1035 psia. The individual considerations evaluated for power uprate included the following: the regulatory basis; the event scenario; condensate inventory and reactor coolant inventory; station battery load; compressed air supply; and loss of ventilation to the control room, reactor protection system rooms and switchgear rooms, HPCS pump and auxiliary rooms, RCIC room, containment, suppression pool and spent fuel pool. The SBO analysis is acceptable to the NRC staff.

3.6 Containment Evaluation

The NMP-2 updated safety analysis report (USAR) provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with power uprate changes some of the conditions for the containment analyses. Section 5.10.2 of Topical Report NEDC-31897, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate" requires the power uprate applicant to show acceptability of the uprated power level for: (1) containment pressures and temperatures, (2) LOCA containment dynamic loads, and (3) safety-relief valve dynamic loads. Appendix G of NEDC-31897 prescribes the approach to be used by power uprate applicants for performing required plant-specific analyses. NMPC did the necessary analyses and discussed the results in the application.

Appendix G of NEDC-31897 states that the applicant will analyze short-term containment responses using the staff-approved M3CPT code. M3CPT is used to analyze the period from when the break begins to when pool cooling begins. M3CPT generates data on the response of containment pressure and temperature (Section 3.6.1), dynamic loads analyses (Section 3.6.2) and for equipment qualification analyses (Section 3.13).

Appendix G of NEDC-31897 also states that the applicant will perform long-term containment heatup (suppression pool temperature) analyses for the limiting safety analysis report events to show that the pool temperatures will remain within limits for:

Containment design temperature,
local pool temperature,
Net positive suction head (NPSH),
pump seals, piping design temperature, and other limits

These analyses will use the SHEX code and ANS 5.1-1979 decay heat assumptions consistent with the NRC staff's letter to Gary L. Sozzi of July 13, 1993. SHEX, which is partially based on M3CPT, is a long term code to analyze the period from when the break begins until after peak pool heatup.

3.6.1 Containment Pressure and Temperature Response

Short-term and long-term containment analyses of containment pressure and temperature response following a large break inside the drywell are documented in the USAR. The short-term analysis is performed primarily to determine the peak drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment after a design basis accident (DBA) LOCA. The long-term analysis is performed primarily to determine the peak pool temperature response.

3.6.1.1 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

NMPC indicated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA at 102 percent of the uprated power using the SHEX code and ANS 5.1 decay heat assumptions prescribed by NEDC-31897. The analyses have been performed using the more realistic RHR pool cooling capability than that which was used in the original analyses (K-factor=240.2 Btu/sec-° F vs. 199.2 Btu/sec-° F), but also with a higher service water temperature (82 °F vs 77 °F). The NRC staff has approved the use of the higher K factor and service water temperature in a safety evaluation to Amendment No. 3 dated April 11, 1988. All other key input parameters for power uprate analyses were essentially the same as those for the original analyses. For the power uprate, the DBA-LOCA peak suppression pool temperature was calculated to be 207.9 °F. This temperature is approximately 1 °F higher than the value given in the USAR but is within suppression pool design temperature limit of 212 °F and meets the ECCS pumps NPSH requirements.

NMPC indicated that the long-term bulk pool temperature response was also evaluated for the non-LOCA limiting event which assumes reactor isolation with only one RHR heat exchanger available to accommodate SRV discharge to the suppression pool. The peak bulk suppression pool temperature calculated with 102 percent of the uprated power was 210.9 °F. This temperature is approximately 2 °F higher than the value obtained with the current power but is within the suppression pool design value of 212 °F.

Based on the results of these analyses, the NRC staff concludes that the bulk suppression pool temperature response remains acceptable after power uprate.

(2) Local Pool Temperature with SRV Discharge

A local pool temperature limit for SRV discharge is specified in NUREG-0783 because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. NMPC indicated that since the NMP-2 has quenchers, no evaluation of this limit is considered necessary. Elimination of this limit for plants with quenchers on the SRV discharge lines is justified in GE Report NEDO-30832, "Elimination of Limits on Local Suppression Pool Temperature for SRV Discharge with quenchers." NEDO-30832 has been evaluated and approved by the NRC staff (SE dated August 29, 1994). However, the local pool temperature has been evaluated at uprated power, and was found to be acceptable with respect to NUREG-0783 limit.

Based on the above, the NRC staff concludes that the local pool temperature limit will remain acceptable after power uprate.

3.6.1.2 Containment Gas Temperature Response

NMPC indicated that the containment drywell design temperature of 340 °F was determined based on a bounding analysis of the superheated gas temperature which can be reached with blowdown of steam to the drywell during a LOCA. The expansion of the reactor steam under these conditions will result in a calculated peak drywell temperature of 325.8 °F. Assuming that there is a 6-hour cooldown period required to completely depressurize the reactor vessel based on a controlled 100 °F/hr cooldown rate, the small steamline break analysis shows the peak value to be approximately 270 °F at current power. Small steamline breaks in the drywell impose the most severe drywell temperature conditions. The changes in the reactor vessel conditions with power uprate will increase the calculated long-term peak drywell gas temperature response during a small-break LOCA by a maximum of a few degrees but will not exceed the drywell design value of 340 °F. Therefore, the drywell gas temperature response after power uprate will remain below the containment design temperature of 340 °F.

NMPC indicated that the wetwell gas space peak temperature response was calculated assuming thermal equilibrium between the pool and wetwell gas space. The reanalysis has shown that the maximum bulk pool temperature will reach 207.9 °F after LOCA and 210.9 °F after alternate shutdown due to power uprate. Therefore, the maximum wetwell gas space temperature due to power uprate will remain below the wetwell design temperature of 270 °F.

Based on its review, the NRC staff concludes that the containment drywell and wetwell gas temperature response will remain acceptable after power uprate.

3.6.1.3 Short Term Containment Pressure Response

NMPC indicated that the short-term containment response analyses were performed for the limiting DBA-LOCA, which assumes a double-ended guillotine break of a recirculation suction line to demonstrate that power uprate operation will not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and differential pressure between the drywell and wetwell occur. These analyses were performed at 102% of the uprated power level, using the GE M3CPT computer code. The reanalysis predicted a maximum containment pressure of 36.8 psig which remains below the containment design pressure of 45 psig. The reanalysis also predicted a maximum drywell-to-wetwell pressure difference of 16.3 psid which remains below the design limit of 25 psid.

TSs definitions, limiting conditions for operation, surveillance requirements and bases relating to the current 39.75 psig value of P_a will not be revised as it remains higher than the maximum containment pressure of 36.8 psig calculated for the power uprate.

Based on its review, the NRC staff concludes that the containment pressure response following a postulated LOCA will remain acceptable after power uprate.

3.6.1.4 Steam Bypass Case

NMPC indicated that the steam bypass of the suppression pool due to a leakage between the drywell and the wetwell airspace during a LOCA event was analyzed to ensure that the primary containment design pressure of 45 psig is not exceeded. The amount of steam bypass leakage is determined by the magnitude and duration of the pressure difference between the drywell and the wetwell during a LOCA (governed by the vent submergence), and by the leakage flow area. These parameters are not affected by reactor power. The assumed time of 30 minutes required for the operator to initiate containment spray operation is not changed. Power uprate will only influence the suppression pool temperature, and subsequently, the primary containment pressure. A bounding evaluation estimated an increase of approximately 0.2 psi in the peak drywell pressure based on the increase in the bulk suppression pool temperature prior to initiation of containment sprays at 30 minutes. The 0.2 psi increase in the peak primary containment pressure due to power uprate will not result in a peak primary containment pressure which exceeds the design value of 45 psig. Assuming the 0.2 psi increase in the peak drywell pressure, the maximum allowable ($A/K^{0.5}$) steam bypass capacity is reduced from 0.057 sq. ft. to about 0.056 sq. ft. (USAR Figure 6.2-28) but remains above the 0.054 sq. ft. value used as the basis for the current TS for allowable bypass leakage. The evaluation shows that the power uprate has negligible impact on the suppression pool steam bypass effects.

Based on the above, the NRC staff concludes that the steam bypass response will remain acceptable after power uprate.

3.6.2 Containment Dynamic Loads

3.6.2.1 LOCA Containment Dynamic Loads

NEDC-31897 requires that the power uprate applicant determine if the containment pressure, temperature and vent flow conditions, calculated with the M3CPT code for power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads were based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by power uprate and thus do not require further analysis.

NMPC indicated that the LOCA dynamic loads which are considered in the power uprate evaluation include pool swell, condensation oscillation (CO), and chugging. The initial drywell pressurization rate used to define the pool swell load bounds the value calculated with the uprated power. The short-term containment response conditions for vent flow rate and pool temperature with power uprate are within the range of test conditions used to define the CO loads. The containment conditions with power uprate in which chugging would occur are within the range of test conditions used to define the chugging loads. Therefore, the LOCA dynamic loads for NMP-2 are not impacted by power uprate.

Based on the above, the NRC staff concludes that the LOCA containment dynamic loads will remain acceptable after power uprate.

3.6.2.2 Safety Relief Valve (SRV) Containment Dynamic Loads

The SRV containment dynamic loads include discharge line loads, pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by SRV opening setpoints, discharge line configuration and suppression pool configuration. Of these parameters only the SRV setpoint is affected by power uprate. NEDC-31897 states that if the SRV setpoints are increased, the power uprate applicant will attempt to show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

NMPC indicated that the highest SRV opening setpoint with power uprate will be 1241 psig. The SRV setpoint which was the basis for the SRVDL loads and the SRV loads on the suppression pool boundary and submerged structures is 1261 psig. Since the highest setpoint with power uprate remains lower than the setpoint used to define the SRV loads, power uprate does not impact the SRV definitions for the first actuation of SRVs. The water leg prior to SRV opening used to define the subsequent actuation loads conservatively assumed the maximum calculated SRVDL reflood height. This is not impacted by power uprate. Therefore, there will be no effect of power uprate on the water leg prior to SRV opening and no impact of power uprate on the subsequent actuation loads. The SRV containment dynamic loads will remain below their original design values after power uprate.

Based on the above, the NRC staff concludes that the SRV containment dynamic loads will remain acceptable after power uprate.

3.6.2.3 Subcompartment Pressurization

NMPC indicated that the design loads on the annulus between the biological shield wall and vessel and the drywell head due to a postulated pipe break in the annulus were evaluated for the limiting subcompartment pressurization event at uprated conditions. The values used for the power uprate evaluation at 102% of the uprated power are not significantly changed from the values used for original analysis at 104.3% of current power. The subcompartment pressurization loads are not significantly affected by power uprate and remain acceptable. It is also noted that the NEDC-31897 methodology does not require subcompartment reanalysis. Based on the above, the NRC staff concludes that the subcompartment pressurization effects will remain acceptable after power uprate.

3.6.3 Containment Isolation

The NEDC-31897 methodology does not address a need for reanalysis of the isolation system. The system designs for containment isolation are not affected by power uprate. The capability of the actuation devices to perform with uprated pressure and flow will comply for acceptability in response to

Generic Letter 89-10 at uprated conditions. Based on its review, the NRC staff finds that the operation of the plant at uprated power level will not impact the containment isolation system.

3.6.4 Post-LOCA Combustible Gas Control

NMPC indicated that the hydrogen recombiners are provided to maintain the containment atmosphere as a non-combustible mixture after DBA-LOCA. The combustibility of the post-LOCA containment atmosphere is controlled by the concentration of oxygen. As a result of power uprate, the post-LOCA production of oxygen and hydrogen by radiolysis will increase proportionally with power level. The original evaluation of the system was performed at 3467 MWt, the evaluation at uprated operation increases only by 2%. Sufficient capacity exists in the combustible gas control system to accommodate the slightly increased oxygen and hydrogen production. Also, recombiner operation is controlled procedurally based on gas concentration in the containment. Based on its review, the NRC staff concludes that the post-LOCA combustible gas control will remain acceptable after uprated power.

3.7 Standby Gas Treatment System (SGTS)

The SGTS is designed to minimize offsite dose rates during venting and purging of both the primary and secondary containment atmosphere under accident or abnormal conditions, while containing airborne particulate and halogens that might be present. The SGTS consists of two identical, parallel, physically separated, 100-percent capacity air filtration assemblies with associated piping, valves, controls, and centrifugal exhaust fans. Effluents from the SGTS connect to a common exhaust line discharging to the exhaust tunnel leading to the main tack. The SGTS draws air from the reactor building.

Following a postulated accident, the SGTS is started, taking over from the normal ventilation system which has been maintaining secondary containment at a slightly negative pressure, ≤ -0.25 inch water gauge (WG). Maintaining this negative pressure serves to prevent unfiltered release of radioactive material from the secondary containment to the environment. During the transfer to SGTS operation, pressure rises momentarily until the SGTS, together with the Category I unit coolers, reestablishes pressure ≤ -0.25 inch WG.

NMPC indicated that appropriate differential temperature requirements will be maintained for uprated operation to ensure that the secondary containment atmosphere temperature is sufficiently above the available service water temperature so that the negative pressure is restored within the time period assumed in the radiological evaluations. The air-flow capacity of the SGTS was selected to accommodate the in-leakage equivalent to one secondary containment air volume change per day and thereby maintain the reactor building at the desired negative pressure. The SGTS capability remains adequate for uprated operation in conjunction with appropriate differential temperature requirements.

NMPC also indicated that the charcoal filter beds are not significantly affected by uprated power level operation. The SGTS is designed to be in compliance with RG 1.52 (Rev. 2) with numerous minor exceptions, including charcoal loading capacity. The SGTS is designed for a charcoal loading capacity of 10mgI/gC as compared to a value of 2.5mgI/gC per RG 1.52 (Rev. 2), and meets the design requirements for 30-day and 100-day LOCA scenarios. The total post-LOCA iodine loading increases less than 4.3% at the uprated conditions and remains within the 10mgI/gC loading limit of the system.

The NRC staff reviewed NMPC's use of 10mgI/gC loading capacity. This exception along with numerous other exceptions to RG 1.52 was submitted to the NRC staff in the FSAR prior to issuance of the NMP-2 operating license. The NRC staff's safety evaluation accepted all exceptions to RG 1.52 but did not discuss the basis for acceptance. The only exception of concern to the NRC staff for power uprate was the charcoal loading capacity.

NMPC provided additional justification for the deviation to the charcoal loading capacity recommendation in RG 1.52 in their letter dated September 16, 1994. NMPC states in their letter that adsorbed iodine in the charcoal would not generate heat at a sufficient rate to result in either combustion of the charcoal or temperatures high enough to cause significant desorption of the iodines. The charcoal adsorption capacity of 10 mgI/gC is within the adsorption capacity of the activated carbon used in the SGTS with respect to loading capacity and adsorption efficiency. The carbon capacity is supported by surveillance test data. The maximum decay heat generation rate for an assumed total charcoal iodine loading of 10 mgI/gC at the power uprate condition for an SGTS train has been calculated to be approximately 15,000 BTU/hr and would occur approximately 250 hours into a design basis LOCA. This maximum heat generation for an operating SGTS train is easily dissipated by the operation of the train's associated fan. Therefore, the NRC staff finds that NMPC's use of 10mgI/gC loading capacity is acceptable.

Based on its review, the NRC staff concludes that the uprated power level operation will not have any impact on the ability of the SGTS to meet its design objectives.

3.8 Fuel Pool Cooling System

The spent fuel pool cooling system is designed to remove the decay heat released from the stored spent fuel assemblies and maintain a pool water temperature at or below 125 °F under normal operating conditions and below a maximum fuel pool design temperature of 150 °F under all other conditions. Backup or supplemental cooling may be provided by the residual heat removal (RHR) system.

As a result of operation at the uprated power level, each reload will affect the decay heat generation in the spent fuel discharged from the reactor and the spent fuel heat load will increase slightly. NMPC's refueling cycle analysis indicated that maximum normal pool heat load of 14.4×10^6 Btu/hr is still within the heat removal capability of just one of the two fuel pool cooling loops at 15 x 10^6 Btu/hr. Thus, operation at uprated power level will not have any negative effect on the cooling capability to keep the fuel pool temperature at or below the design temperature and maintain adequate fuel pool cooling for normal discharge (offload) conditions.

The full core offload condition may cause the heat load in the spent fuel pool to reach a new maximum at 31.3 x 10^6 Btu/hr. If the actual heat load due to full core offload is higher than the total design capacity of the two fuel pool heat exchangers (30 x 10^6 Btu/hr), the residual heat removal (RHR) system can adequately provide 1.3 x 10^6 Btu/hr in additional spent fuel pool cooling. Therefore, NMPC concluded that operation at uprated power level will not have any negative effect on the capability to maintain adequate spent fuel pool cooling for full core discharge conditions.

An issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," October 7, 1993, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the licensee under separate cover.

Based on its review, the NRC staff agrees with NMPC that operation at uprated power will not prevent the spent fuel pool cooling system from performing its design function.

3.9 Water Systems

NMPC evaluated the impact of power uprate on the various plant water systems. The systems analyzed below are as follows: service water systems, main condenser, circulating water system, normal heat sink, reactor building closed cooling water system, and turbine building closed cooling water system. In addition, discharge limits for various parameters were analyzed.

3.9.1 Service Water System

The NRC staff evaluation of the service water system is divided into safety-related loads and nonsafety-related loads.

3.9.1.1 Safety-Related Loads

The safety-related service water system is designed to provide a reliable supply of cooling water during and following a design basis accident for the following systems.

3.9.1.1.1 Emergency Equipment Service Water System

NMPC indicated that safety-related performance of the emergency equipment service water (EESW) system during and following the most demanding design basis event, the LOCA, is not significantly dependent on reactor rated power for the following equipment and systems: emergency diesel generator coolers, control building chilled water chillers, RHR pump seal coolers, DBA hydrogen recombiners, reactor building ventilation recirculation cooling coils, reactor building coolers, control building coolers, diesel generator building coolers, service water pump bay unit coolers, and spent fuel pool emergency makeup.

The diesel generator loads and the RHR system flows remain unchanged for LOCA conditions following uprated operation. The building cooling loads remain essentially the same as for uprated power level operation because the equipment performance in these areas is not significantly changed for post-LOCA conditions. Additionally, the ability to supply emergency makeup to the spent fuel pool is also unchanged since uprated power level operation does not require the modification of the service water system.

Based on its review, the NRC staff agrees with NMPC that operation at uprated power level will have minimal impact on the EESW system operation.

3.9.1.1.2 Residual Heat Removal Service Water System

NMPC indicated that the power uprate will not increase the cooling requirements for the residual heat removal (RHR) system and its associated Service Water System.

Based on its review, the NRC staff agrees with NMPC that uprated power level operation will not have a significant impact on the RHRSW system.

3.9.1.2 Nonsafety-Related Loads

The normal service water (SW) system is designed to supply cooling water to the closed cooling water systems and other auxiliary heat loads. The major service water heat load increases from power uprate reflect an increase in main generator losses rejected to the stator water coolers, hydrogen coolers and exciter coolers in addition to increased bus cooler heat loads. NMPC indicated that even though this increase in service water heat loads due to uprated power level operation is projected to be approximately proportional to the uprate itself, the SW system is adequate to remove the additional heat loads.

Since the SW system does not perform any safety function, the NRC staff has not reviewed the impact of the uprated power level operation to the SW system design and performance.

3.9.2 Main Condenser/Circulating Water/Normal Heat Sink Performance

The main condenser, circulating water, and normal heat sink (cooling tower) systems are designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. NMPC indicated that performance of the main condenser, circulating water, and the cooling tower were evaluated for power uprate and determined that the systems are adequate for uprated power level operation.

Since the main condenser, circulating water, and normal cooling tower systems do not perform any safety function, the NRC staff has not reviewed the impact of the uprated power level operation on the designs and performances of these systems.

3.9.3 Reactor Building Closed Cooling Water System

The reactor building closed cooling water (RBCCW) system is designed to cool various auxiliary equipment in the reactor building during normal plant operations. NMPC indicated that the increase in heat load due to uprated power level operation does not significantly impact the capability of the RBCCW system to perform its intended function.

Since the RBCCW system does not perform any safety function, the NRC staff has not reviewed the impact of the uprated power level operation to the RBCCW system design and performance.

3.9.4 Turbine Building Closed Cooling Water System

The turbine building closed cooling water (TBCCW) system supplies cooling water to auxiliary plant equipment in the turbine building. NMPC indicated that even though the heat-load increase on the TBCCW system due to power uprate are those related to the operation of the turbine-generator, the system contains sufficient capacity to assure that adequate heat removal capability is available for uprated power level conditions.

Since the TBCCW system does not perform any safety function, the NRC staff has not reviewed the impact of the uprated power level operation to the TBCCW system design and performance.

3.9.5 Discharge Limits

NMPC compared the current effluent discharge limits (to water) to observed discharges and realistic and bounding analysis discharges for power uprate. These discharge limits include net heat addition, discharge temperature, intake/discharge delta temperature, chlorine concentration, and flow rate. The comparison demonstrates that the plant will remain within the State of New York discharge limits during operation at uprated power level.

NMPC indicated that the power uprate will not require any changes to environmental discharge limitations as they apply to current unit operation. That is, none of the present limits for plant environmental releases such as service water discharge temperature or plant vent radiological limits will be increased as a consequence of uprated power level operation. In the unlikely situation that plant releases approach environmental limits, plant operation will be managed such that the existing limits would not be violated. However, it is not expected that any of the existing environmental limits will be approached.

Based on its review, the NRC staff agrees with NMPC that uprated power level operation will not have a significant impact on the effluent discharge limits.

3.9.6 Ultimate Heat Sink

The ultimate heat sink (UHS) for NMP-2 is Lake Ontario. NMPC has not requested any changes to the normal operational discharge limits to the UHS. NMPC indicated that accident mitigation has been shown to be acceptable assuming the same maximum service water temperature (82 °F) to be available from the lake. Therefore, the UHS will be adequate for uprated power level operation.

Based on its review, the NRC staff agrees with NMPC's conclusion that the UHS design is acceptable for the uprated power level operation and no modification to the UHS system is required.

3.10 Power-Dependent Heating, Ventilation, and Air-Conditioning

The Heating, Ventilation, and Air-Conditioning (HVAC) systems consist mainly of heating, cooling supply, exhaust and recirculation units in the turbine building, reactor building, and the drywell. Uprated power level operation is expected to result in slightly higher process temperatures and a small increase in the heat load due to higher electrical currents in some motors and cables.

The areas most affected by operation at uprated power level will be drywell, main steam tunnel, and heater bay areas in the turbine building.

Specifically, the heat loads are expected to increase about 3% in the drywell, about 1% in the main steam tunnel, and about 6% in the heater bay area. Based on samples of plant operating data, these increases are within the excess design capability available for the HVAC systems. Thus, the design of the HVAC systems is not adversely affected by power uprate.

Based on its review, the NRC staff agrees with NMPC that uprated power level operation will not have a significant impact on the plant power-dependent HVAC systems.

3.11 Fire Protection

NMPC indicated that the operation of the plant at the uprated power level would not adversely affect the fire suppression or detection systems. There are no physical plant configuration or combustible load changes resulting from the uprated power level operation. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change and are adequate for the uprated power level conditions. The operator actions required to mitigate the consequences of a fire are not adversely affected. Therefore, the fire protection systems and analyses are not adversely affected by uprated power level operation.

Based on its review, the NRC staff finds that the fire suppression and detection systems are not affected by the power uprate.

3.12 Postulated Pipe Breaks

3.12.1 High Energy Line Break

The slight increase in the operating pressure and temperature caused by the power uprate results in a small increase in the mass and energy release rates following a high-energy line break (HELB). Evaluation of HELB outside the primary containment at the uprated power level showed that there is no change in relative humidity and the original mass and energy blowdown rate was shown to be bounding or insignificantly affected; therefore, the resulting pressure/temperature profiles are not significantly changed from the existing profiles.

NMPC has reevaluated the HELB for the main steam system, the feedwater system, the high pressure ECCS, the reactor core isolation cooling system, the reactor water cleanup system, and the control rod drive system. As a result of this evaluation, NMPC has concluded that the affected building and cubicles that support the safety-related functions are designed to withstand the resulting pressure and thermal loading following a HELB. The NRC staff has reviewed the results of NMPC's reanalysis and finds them acceptable.

NMPC has also evaluated the calculations supporting the disposition of potential targets of pipe whip and jet impingement from the postulated HELBs and determined that they are adequate for the safe shutdown effects in the uprated power condition. Existing pipe whip restraints and jet impingement shields and their supporting structures have also been determined to be adequate for operation at uprated power.

Based on its review, the NRC staff concludes that the analyses for HELBs outside containment are acceptable for the proposed operation at the uprated power level.

3.12.2 Moderate Energy Line Crack

NMPC determined that uprated power level operation has no impact on the moderate energy line crack. Based on a review of the high pressure ECCS, the reactor core isolation cooling system, the reactor water cleanup system, and the control rod drive system, NMPC concluded that the original moderate energy line crack analysis is not affected by operation at the uprated power level.

Based on its review, the NRC staff agrees with NMPC that uprated power level operation has no impact on the moderate energy line crack.

3.13 Equipment Qualification (EQ)

NMPC's July 22, 1993, submittal was supplemented on April 10, 1994, tc provide additional details of analyses of the effect of the power uprate on equipment qualification. The NRC staff evaluation and conclusions follow.

3.13.1 EQ of Electrical Equipment

NMPC has evaluated safety-related electrical equipment to assure qualification for the normal and accident conditions expected in the area where the devices are located and that conservatisms have been applied to demonstrate that all components are qualified for safety function generation at uprated power level conditions. The results of their evaluation indicates that the slight increase (1.36%) in radiation dose will not affect previously defined radiation qualification lifetimes, and that accident thermal and pressure considerations remain unchanged. Normal temperatures will increase slightly due to an increase in operating dome pressure, the effects of which are discussed below. No replacement or modification of any equipment is required due to the uprated power conditions.

3.13.1.1 Inside Containment

The EQ for safety-related electrical equipment located inside the containment is based on main steamline break or Design Basis Accident - loss of coolant accident (DBA/LOCA) conditions and their resultant temperature, pressure, humidity, dynamic loads, and radiation consequences. The EQ for equipment inside containment also includes consideration of the environments expected to exist during normal plant operation.

NMPC, in their reevaluation of the equipment qualification for the uprated power level conditions, determined that all equipment is bounded from the viewpoint of post-accident pressure, temperature, humidity, and dynamic loads. A small number of components were impacted by the higher normal operating temperatures that are due to uprated power level conditions, resulting in reduced qualification lifetimes. NMPC modified the preventative maintenance program to assure replacement of the affected components before the end of their qualified lifetimes.

Based on its review, the NRC staff finds NMPC's approach to qualification of electrical equipment inside containment acceptable.

3.13.1.2 Outside Containment

The EQ for equipment outside containment uses the harsh, accident portions of the temperature, pressure, and humidity environments which result from a steam line break (e.g., in the pipe tunnel) or other high energy line breaks, whichever is limiting for each plant area. The EQ for equipment outside containment also includes consideration of the environments expected to exist during normal plant operation.

NMPC, in their reevaluation of the equipment qualification for the uprated power level conditions, determined that all equipment is bounded from the viewpoint of post-accident pressure, temperature, humidity, and dynamic loads. A small number of components were impacted by the higher normal operating temperatures that are due to uprated power level conditions, resulting in reduced qualification lifetimes. NMPC has modified the preventative maintenance program to ensure replacement of the affected components before the expiration of their qualified lifetimes.

Based on its review, the NRC staff finds NMPC's approach to qualification of electrical equipment outside containment acceptable.

3.13.2 EQ of Non-Metallic Components of Mechanical Equipment

NMPC determined that all non-metallic components of mechanical equipment are bounded from the viewpoint of post-accident pressure, temperature, humidity, and dynamic loads. A small number of components were impacted by the higher normal operating temperatures that are due to uprated power level conditions. The qualification lifetimes of these components have been reduced, and the preventive maintenance program was modified to ensure replacement of the affected components before the expiration of their qualified lifetimes.

Based on its review, the NRC staff finds NMPC's approach to qualification of non-metallic components of mechanical equipment acceptable.

3.13.3 Mechanical Component Design Qualification

NMPC indicated that the mechanical design of equipment/components (e.g., pumps, heat exchangers, etc.) is affected by operation at the uprated power level due to slightly increased temperatures, pressure, and flow. However, the uprated power operating conditions do not significantly affect the cumulative usage fatigue factor of mechanical components.

Increases to component nozzle loads and component support loads due to the uprated power level conditions were evaluated with the Nuclear Steam Supply System (NSSS) and the Balance-of-Plant piping assessment. It was shown that thermal and vibration displacement limits for hangers and snubbers due to power uprate conditions are within allowable limits and load increases for other supports such as anchors, guides and penetrations, and reactor pressure vessel nozzles are acceptable. All of the evaluated stresses and cumulative fatigue usage factors were shown to be within American Society of Mechanical Engineers Code allowable limits. These components have been evaluated to have adequate capability for operation at the uprated power level.

Based on its review, the NRC staff agrees with NMPC that operation at the uprated power level will not have a significant impact on the above system.

3.14 Instrumentation and Control

Many of these TS changes involve changes to the Reactor Protection System trip and interlock setpoints. These changes are intended to maintain the same margin between the new operating conditions and the new trip points as existed before the proposed power uprate.

The conservative design calculations for the initial licensing of NMP-2 resulted in setpoints which provided excess reactor coolant flow capacity and corresponding margins in the power conversion system. For NMP-2, these margins (e.g. 5% rated steam flow) result in the capability to increase the core operating power level by approximately 4.3% This safety evaluation is limited to setpoint changes for the identified instrumentation and is predicated on the assumption that the analytical limits used by NMPC are based on application of approved design codes.

The following setpoint changes have been proposed by NMPC:

1. Reactor Vessel Pressure High Scram

   Change trip from ≤ 1037 psig to ≤ 1052 psig.
   Change Analytical Limit from ≤ 1057 psig to ≤ 1072 psig.

2. Main Steam High Flow

   The analytical limit for main steam high flow is based on the 140% of the uprated steam flow condition.
   Change trip from ≤ 103 psid to ≤ 121.5 psid.
   Change Allowable Value from ≤ 109.5 psid to ≤ 122.8 psid.

3. Turbine First-Stage Scram Bypass Pressure

   The turbine first stage pressure setpoint was changed to reflect the expected pressure at the new 30% power point.
   Change bypass setpoint from ≤ 119 psig to ≤ 125.8 p[illegible]g.
   Change Allowable Value from ≤ 129.6 psig to ≤ 135[illegible] psig.

4. ATWS Recirculation Pump Trip Reactor Vessel Pressure - High

   Change trip setpoint from ≤ 1050 psig to ≤ 1065 psig.
   Change Allowable Value from ≤ 1065 psig to ≤ 1080 psig.

5. Main Steam Line Tunnel Temperature

The main steam line tunnel temperature trip setpoints were changed to reflect the increase in the operating temperature.
Change trip setpoint for high temperature from ≤ 165.7 °F to ≤ 167.2.
Change Allowable Value for high temperature from ≤ 169.9 °F to ≤ 170.6 °F.

Change trip setpoint for ΔT high from ≤ 66.7 °F to ≤ 70 °F
Change Allowable Value for ΔT high from ≤ 71.3 °F to ≤ 71.7 °F.
Change trip setpoint for MSL Lead Enclosure high temperature from ≤ 146.7 °F to ≤ 148.2 °F.
Change Allowable Value for MSL Lead Enclosure high temperature from ≤ 150.9 ° to ≤ 151.6 °F.

NMPC's submittal dated July 22, 1993 and December 2, 1994, did not provide information regarding the methodology used for instrument setpoint calculations. Therefore, by letter dated February 24, 1995, the NRC staff requested additional information regarding instrument setpoint methodology. NMPC, by letter dated March 8, 1995, provided responses to the NRC staff's request and confirmed that GE Licensing Topical Report NEDC-31336 was used for instrument setpoint calculations. The NRC staff previously reviewed this topical report and accepted it with some minor exceptions. These exceptions are under NRC staff review and will be resolved on a generic basis. They do not affect the NRC staff's evaluation of the proposed NMP-2 changes at this time. NMPC in their letter also confirmed that the calculation methodology is identical to the plants which have been reviewed and approved by the NRC staff previously, e.g. Fermi-2 and WNP-2.

The proposed setpoint changes are intended to maintain the existing margins between operating conditions and the reactor trip setpoints. Thus, margins to the new safety limits will remain the same as the current margins. These new setpoints also do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not affected.

Based on the above, the NRC staff concludes that NMPC's instrument setpoint methodology and the resulting setpoint changes incorporated in the TS for the power uprate are consistent with the NMP-2 licensing basis and are, therefore, acceptable.

3.15 Radiation Levels

NMPC evaluated the effects of power uprate on radiation levels in the NMP-2 facility during normal and anticipated operational occurrences, as well as from postulated accident conditions. NMPC concluded that radiation levels from both normal and accident conditions may increase slightly upon power uprate. For example, normal operational radiation levels in most of the plant are expected to increase by no more than the increase in power level (4.3%). In a few areas near the reactor water piping and liquid radwaste equipment, the radiation levels could increase to 9.5 percent.

However, any such increase is bounded by conservatism in the original design and analysis. Also, individual exposures to plant workers will be maintained within acceptable limits by the existing ALARA program, which controls access to radiation areas. Procedural controls could compensate for such slightly increased radiation levels.

The offsite doses associated with normal operation are not significantly affected by operation at the uprated power level, and should remain below the limits of 10 CFR Part 20 and Appendix I to 10 CFR Part 50.

On the basis of its review, the NRC staff concludes that no significant adverse effect or increase in radiation levels will result onsite or offsite from the planned power uprate.

3.16 Radiological Consequences - Design Basis Accidents

NMPC stated that the original radiological consequence analyses could not be exactly reconstituted and, therefore, the reconstituted analyses were performed using methodology described in the UFSAR with the original licensing basis assumption at 3489 MWt (105% of current power level). NMPC's reconstituted analyses indicate that the calculated offsite radiological consequences [illegible]es are within the dose reference values stated in 10 CFR Part 100 and [illegible] comply with the dose limits to control room operators given in General Design Criterion (GDC) 19.

In its NMP-2 safety evaluation issued in February 1985, the NRC staff analyzed radiological consequences at 3489 MWt (105% of current power level). The events evaluated for uprate were the LOCA, the fuel handling accident (FHA) and the control rod drop accident (CRDA). Whole body and thyroid dose were calculated for the exclusion area boundary (EAB), the low population zone (LPZ), and the control room. The plant-specific results for the power uprate remain well below established regulatory limits. The doses resulting from the accidents analyzed are compared below with the applicable dose limits.

TABLE 1 - LOCA Radiological Consequences

	UFSAR 3489 MWt (rem)	SER 3489 MWt (rem)	Part 100 Limits
EAB:			
Whole Body Dose	6.3	2 6	25
Thyroid Dose	232.0	224 0	300
LPZ:			

Whole Body Dose	1.9	2.4	25
Thyroid Dose	56.0	292.0	300

TABLE 2 - FHA Radiological Consequences

	UFSAR 3489 MWt (rem)	SER 3489 MWt (rem)	Part 100 Limits
EAB:			
Whole Bo'v Dose	0.64	0.27	6
Thyroid Dose	44.00	38.00	75
LPZ:			
Whole Body Dose	0.16	0.030	6
Thyroid Dose	9.30	4.900	75

TABLE 3 - CRDA Radiological Consequences

	UFSAR 3489 MWt (rem)	SER 3489 MWt (rem)	Part 100 Limits
EAB:			
Whole Body Dose	0.02	0.04	6
Thyroid Dose	0.0(3	0.40	75
LPZ:			
Whole Body Dose	0.0051	0.01	6
Thyroid Dose	0.1760	0.30	75

The preceding analysis was based on 105 percent of current power, i.e. approximately equivalent to the uprated power level of 3467 Mwt, using methodologies currently approved by the NRC. After reviewing the information submitted by NMPC, the NRC staff concludes that despite the power uprate the analyzed consequences of postulated accidents will remain within the limits of 10 CFR Part 100 and the GDC 19 dose limit, and are, therefore, acceptable.

NMPC also evaluated main control room (MCR) habitability, confirming that post-accident MCR and Technical Support Center (TSC) doses remained within the limits of GDC 19 of 10 CFR Part 50, Appendix A.

On the basis of its review of NMPC's major assumptions, the methodology used in NMPC reconstituted dose calculations, and the NRC staff's original safety evaluation, the NRC staff finds that the offsite radiological consequences and control room operator doses at the uprated power level of 3467 MWt will continue to remain below 10 CFR Part 100 dose reference values and the GDC 19 dose limit, and therefore, are acceptable.

3.17 Structural Integrity of Vessel, Piping, and Equipment

In a letter dated January 3, 1995 (Reference 5), NMPC responded to the NRC staff's November 21, 1994, request for additional information regarding various aspects associated with the NMP-2 power uprate that may differ from those in the GE generic evaluation for BWR power uprate. In the January 3, 1995 letter, NMPC also provided a fatigue evaluation for the power uprate conditions, GE NEDC-32015 dated September 1994. In a letter dated December 2, 1994, NMPC transmitted revised pages reflecting changes to the proposed power uprate submittal and attachments, resulting from various calculations and analyses completed since the July 22, 1993, submittal. The changes are considered minor and do not alter the conclusion of the original submittal regarding the structural integrity of the reactor coolant pressure boundary.

The GE generic guidelines for BWR power uprate effects were based on a 5% higher steam flow, an operating temperature increase of 5 °F and an operating pressure increase of 40 psi. For NMP-2, the maximum reactor vessel dome pressure increases from 1005 psig to 1020 psig, the dome temperature increases from 547 °F to 549 °F and the steam flow rate increases from 14.3×10^6 lb_m/hr to 15.0×10^6 lb_m/hr (approximately a 4.9% increase). The maximum core flow rate will remain unchanged for the NMP-2 power uprate conditions, which is consistent with GE generic guideline assuming no change in core flow.

3.17.1 Reactor Pressure Vessel (RPV) and Internals

NMPC evaluated the reactor vessel and internal components considering load combinations that include reactor internal pressure difference (RIPD), LOCA, safety relief valve (SRV) discharge, and seismic and fuel lift loads, as defined in the NMP-2 Updated Final Safety Analysis Report (UFSAR).

NMPC evaluated LOCA loads such as pool swell, CO, and chugging for the NMP-2 power uprate and found that the test conditions used to define NMP-2 design basis LOCA dynamic loads are bounding for the uprated power conditions with respect to drywell and wetwell pressure, vent flow rate, and suppression pool water temperature. The design basis SRV containment dynamic loads that affect the reactor vessel and piping systems are defined based on an SRV opening setpoint pressure of 1261 psig which is greater than the highest setpoint pressure of 1241 psig for the power uprate. Therefore, the NMP-2 SRV dynamic loads are not impacted by the power uprate. The potential fuel lift loads are affected by the scram uplift force and reactor building upward motion due to seismic and hydrodynamic loads such as LOCA and SRV loads. These loads are not significantly impacted by the power uprate. Therefore, the NRC staff concurs with NMPC's conclusion that the potential increase in fuel lift due to the power uprate is n· ,ıigible. The calculated RIPDs for the uprated power conditions were summarized in Tables 3-1, 3-2 and 3-3 for normal, upset and faulted conditions, respectively.

The stresses and fatigue usage factor for reactor vessel components were evaluated by NMPC in accordance with the requirements of the 1971 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB with Winter 1972 Addenda to assure compliance with the NMP-2 original Code of record. NMPC performed evaluations of critical internal components in Section 3.3 of Reference 2 for the effects of increased RIPDs for all service conditions and found all evaluated internal components to be acceptable for the power uprate. The limiting fatigue usage factor calculated for the uprated power level in GE NEDC-32015 (September, 1994), was 0.965 for the carbon steel section of the feedwater nozzle. No new assumptions were used in the analysis for the power uprate condition.

Based on the NRC staff's review, the maximum stresses and fatigue usage factor as stated by NMPC are within the Code allowable limits and are, therefore, acceptable.

3.17.2 Con. l Rod Drive System

NMPC evaluated the NMP-2 control rod drive mechanism (CRDM) for the uprated power conditions in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with addenda through Winter 1975. The limiting component of the CRDM was identified to be the indicator tube. The maximum calculated stresses were within the ASME Code allowable for the licensing basis load combinations that include a maximum CRDM internal water pressure of 1750 psig and hydrodynamic loads such as LOCA and SRV loads. These loads are not significantly affected by the power uprate at NMP-2. The maximum calculated fatigue usage factor based on ASME Code NB-3222.4 is 0.15 for the CRDM main flange for 40 years of plant operation.

The increase in the reactor dome pressure, operating temperature and steam flow rate as a result of the power uprate are bounded by the conditions assumed in the General Electric generic guidelines for the power uprate. The CRDM was originally evaluated for a normal maximum reactor dome pressure of 1060 psig which is higher than the power uprate dome pressure of 1020 psig. In addition, NMPC indicated that the CRDM has been tested at simulated reactor pressure up to 1250 psig, which bounds the high pressure scram setpoint of 1086 psig for the power uprate.

Based on the above review, the NRC staff concurs with NMPC's determination that the CRDM will continue to meet its design basis and performance requirements at uprated power conditions.

3.17.3 Reactor Coolant Piping

NMPC evaluated the effects of the power uprate conditions, including higher flow rate, temperature and pressure for thermal expansion, fluid transients

3.17.3 Reactor Coolant Piping

NMPC evaluated the effects of the power uprate conditions, including higher flow rate, temperature and pressure for thermal expansion, fluid transients and vibration effects on the reactor coolant pressure boundary (RCPB) and the balance-of-plant (BOP) piping systems, including in-line components such as equipment nozzles, valves and flange connections, and pipe supports. The evaluation of piping systems affected by the power uprate follows the methodology in Appendix K of GE generic guideline, Reference 3. The original code of record as specified in NMP-2 UFSAR and the ASME Code allowables were used and no new assumptions were introduced that were not in the original analyses.

The RCPB piping systems evaluated included the main steam and associated vent and drain lines, reactor recirculation, reactor water clean-up (RWCU), reactor core isolation cooling (RCIC), feedwater, high pressure core spray (HPCS), low pressure core spray (LPCS), residual heat removal (RHR), control rod drive (CRD) and standby liquid control (SLC) lines. NMPC's evaluation of the RCPB piping systems involved an assessment of the maximum increase in stresses for the power uprate condition (due to increase in pressure, temperature and fluid transient loads) against the design margins available in the original design basis analyses, and the performance of stress analyses in accordance with requirements of the Code and the ASME Code Addenda of record under the power uprate conditions. NMPC concluded that the maximum stress levels and fatigue usage factors satisfy the Code requirements for the piping systems evaluated and that power uprate will not have an adverse effect on the reactor coolant piping system design.

The BOP systems evaluation included portions of piping systems listed under Section 3.5 of the submittal and systems that are affected by the power uprate, such as condensate, reactor vessel instrumentation, turbine drains, extraction steam and safety/relief valve discharging. NMPC evaluated the BOP piping systems first by comparing the original design basis conditions with those for the proposed uprated conditions. For those systems whose design temperature and pressure did not envelop the uprate power conditions, NMPC performed stress analyses based on the power uprate conditions, and concluded that the calculated pipe stress levels and fatigue usage factors remained within the allowable Code limits. NMPC indicated in the initial submittal that evaluation of a Class 4 (ANSI B31.1) feedwater piping was not completed at the time the submittal was prepared, but this piping was later evaluated to meet the design limits under the uprated power conditions, as stated by NMPC in its January 3, 1995, letter.

NMPC evaluated pipe supports including anchorage, equipment nozzles, and penetrations by comparing the increased piping interface loads on the system components under the power uprate conditions, with the margin in the original design basis calculation. NMPC concluded that there is sufficient margin and that the evaluated components have adequate capacity for the power uprate. The effect of power uprate conditions on thermal and vibration displacement limits was also evaluated by NMPC for struts, springs and pipe snubbers, and found to be acceptable. NMPC reviewed the original postulated high energy line break (HELB) analysis and concluded that the existing HELB analyses are bounding for the power uprate, and no new pipe break locations were identified.

Based on its review of NMPC's submittal, the NRC staff concludes that the design of piping, components and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant piping and supports in the power uprate conditions.

3.17.4 Equipment Seismic and Dynamic Qualification

Based on the review of the proposed power uprate amendment, the NRC staff finds that the original seismic and dynamic qualification of the safety related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons.

1. Seismic loads are unchanged by power uprate;

2. The original LOCA and SRV load conditions bound the power uprate conditions; and

3. No new pipe break locations will result from the power uprated conditions.

Based on its review, the NRC staff finds that NMPC's proposed power uprate amendment has no significant adverse effect on the structural and pressure boundary integrity of the reactor coolant piping systems, components, and their supports, reactor internals, core support structure, the Control Rod Drive Mechanisms and the BOP piping systems, and is therefore acceptable.

3.18 Human Factors

The NRC staff reviewed the July 22, 1993, submittal and determined the need for additional information concerning changes to the operator interfaces and the emergency operating procedures as a result of the uprate. The NRC staff issued a letter July 26, 1994, requesting additional information. The questions covered the clarification of terms in the submittal and requested information as to whether the power uprate would change the time requirements for operator actions needed for accident mitigation, change procedures, or result in any change in the scope or nature of operator response.

By letter dated August 23, 1994, NMPC responded to the NRC staff's request. The term "time window" was equated with the "window of opportunity" between the time an operator is provided with a cue to take specific action and the time at which the consequences of failing to perform the action are unavoidable. NMPC also stated that the impact on the operator will be minor, primarily resulting from adjustments in the emergency operating procedure threshold cues to conform to the uprated conditions. NMPC stated that there will be no changes to the type or scope of procedures required, no change to the scope or nature of operator responses required, and the power uprate will not significantly change the operator reliability values or overall plant safety measures as calculated by the Independent Plant Evaluation.

Based on the original July 22, 1993, submittal and the information supplied in NMPC's response dated August 23, 1994, the NRC staff has determined that the questions associated with the proposed NMP-2 power uprate have been adequately addressed, and concludes that the power uprate should not adversely affect operator actions or operator reliability.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact have previously been prepared and published in the Federal Register on March 2, 1995 (60 FR 11689). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 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.

7.0 REFERENCES

1. Letter from B. Ralph Sylvia, NMPC, to NRC, "Proposed License Amendment -- Uprated Operation," July 22, 1993.

2. Letter from William T. Russell, NRC, to Patrick W. Marriott, GE, "Staff Position Concerning General Electric Boiling Water Reactor Power Uprate Program," September 30, 1991.

3. GE Licensing Topical Report NEDC-31897P-1, "Generic Guidelines For General Electric Boiling Water Reactor (BWR) Power Uprate," June 1991. (Proprietary information. Not publicly available.) and GE Nuclear Energy, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (nonproprietary), February 1992; and NEDC-31897P-A, Class III (proprietary), May 1992.

4. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (nonproprietary), March 1992; and Supplements 1 and 2.

5. GE Nuclear Energy, "Power Uprate Safety Analysis for NMP-2," Licensing Topical Report NEDC-31994P, Class III, Revision I, (Proprietary), May 1993.

6. GE Nuclear Energy, "NMP-2 SAFER/GESTR-LOCA, Loss-of-Coolant Analysis," Licensing Topical Report NEDC-31830P, Class III (Proprietary), Revision 1, November 1990.

7. Letter to the NRC from B. R. Sylvia, NMPC, response to request for additional information, dated October 6, 1994.

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