

October 20, 1994

MEMORANDUM TO: Robert A. Capra, Project Director (13D-1)
Project Directorate I-1
Division of Reactor Projects I,II

FROM: Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety & Analysis

SUBJECT: NINE MILE POINT UNIT NO. 2 - POWER UPRATE (TAC NO. M87088)

Attachment 1 is the Reactor Systems Branch input to the Safety Evaluation being prepared by your Project Directorate for the subject power uprate license amendment. It is my understanding that you will use this and other technical branch inputs for developing the overall staff safety evaluation for this license amendment. Attachment 2 is our SALP evaluation for this review.

Attachments:
As stated

cc: G. Holahan
D. Brinkman

Contact: R. Frahm, SRXB/DSSA
504-2866

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ATTACHMENT 1

1.0 Introduction

Niagara Mohawk Power Corporation (NMPC), the licensee for Nine Mile Point, Unit No.2 (NMP-2), submitted a request by letter on July 22, 1993 to uprate the licensed power level from 3323 Mwt to 3467 Mwt (Reference 1). This represents approximately a 4.3% increase in thermal power with a 5% increase in rated steam flow. The planned approach to achieve the higher power level consists of (1) an increase in the core thermal power to create an increased steam flow, (2) a corresponding increase in feedwater flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along extension of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in General Electric report NEDC 31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," June 1991 (Reference 2). The operating pressure will be increased approximately 15 psi to assure 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.

2.2 Thermal Limits Assessment

The operating limit 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 3). The MAPLHGR and LHGR limits will also be maintained as described in this reference. The plant specific safety evaluation for NMP-2 is contained in References 4, and 5.

2.3 Reactivity Characteristics

2.3.1 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.

2.4 Stability

Ongoing activities by the BWR Owners' Group and the NRC 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 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, Technical Specifications and associated implementing procedures, as requested by the NRC Bulletin, shall be incorporated by the licensee 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 the licensee 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 staff.

2.5 Reactivity Control

2.6.1 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. The licensee has indicated that CRD performance during power uprate will meet current Technical Specification requirements. The licensee will continue to monitor by various surveillance requirements the scram time performance as required in the plant Technical Specifications 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.

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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.

3.2 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 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 an MSIV closure with a failure of the valve position scram. This transient was analyzed by the license 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). technical specification 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 Technical Specification. Up-rated 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. The licensee's analysis plan is acceptable.

3.4 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 million lbm/hr. The evaluation by the licensee 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. The licensee 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 set point will remain the same. The licensee concluded that the changes due to power uprate and ELLL are small and are bounded by the RRC design basis.

The licensee should perform power uprate start up 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), the licensee 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 staff.

3.7 Main Steam Isolation Valves (MSIVs)

The main steam isolation valves (MSIVs) have been evaluated by the licensee, 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 Technical Specification 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 staff.

3.8 Reactor Core Isolation Cooling System (RCIC)

The reactor core isolation cooling system (RCIC) provides core cooling when the reactor pressure vessel (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 the licensee, 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 the licensee 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. This is acceptable to the staff. The staff requires that the licensee provide assurance that the RCIC system will be capable of injecting its design flow rates at the conditions associated with power uprate. Additionally, the licensee 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.

3.9 Residual Heat Removal System (RHR)

The residual heat removal system (RHR) 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.

3.9.1 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 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 Regulatory Guide 1.139 shows that the reactor can be cooled to 200°F in less than the 36-hour criterion.

3.9.2 Suppression Pool Cooling Mode

The functional design basis for suppression pool cooling mode (SPCM) stated in the 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 the licensee confirms that the pool temperature will stay below its design limit at uprated conditions.

3.9.3 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.

3.10 REACTOR WATER CLEANUP (RWCU) SYSTEM

The RWCU system pressure and temperature will increase slightly as a result of power uprate. The licensee 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.

4.0 ENGINEERED SAFETY FEATURES

4.2 Emergency Core Cooling Systems (ECCS)

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 to the 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 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.

4.2.1 High Pressure Core Spray (HPCS)

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

4.2.2 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 do 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.

4.2.3 Low Pressure Core Spray (LPCS) System

The hardware for the low pressure core spray are 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.

4.2.4 Automatic Depressurization Systems (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCS failure. This function allows low pressure coolant injection (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 the licensee 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.

4.3 ECCS Performance Evaluation

The emergency core cooling systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (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 Appendix K. The General Electric fuel, used in NMP-2 was analyzed by the licensee (Reference 5) with the NRC-approved methods. The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

The licensee used the 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 the licensee 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 Minimum Critical Power Ratio (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 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.

The licensee also reevaluated the ECCS performance for single loop operation (SLO) using the S/G - LOCA methodology. The 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, the licensee claimed that no MAPLHGR reduction is required for SLO. The staff asked the licensee 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. The licensee reviewed this 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. The licensee 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 staff.

The impact of Increased Core Flow (ICF), up to 115 Mlb/hr, 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 uncovering.

9.0 REACTOR SAFETY PERFORMANCE FEATURES

9.1 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. The licensee 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 the licensee 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 Core Operating Limits Report (COLR) developed by the licensee.

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

9.3 SPECIAL EVENTS

9.3.1 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, reactor pressure vessel (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.

9.3.2 Station Blackout (SBO)

The NMP-2 SBO plant responses were evaluated at a steam flow increase of a 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 SRXB staff. However, there are other technical issues that must be evaluated by other branches of the technical staff.

REFERENCES

1. Letter to the NRC from B. Ralph Silva, NMPC, requesting an amendment to the NMP-2 operating license and Technical Specifications to increase licensed power level with Extended Load Line Limit, dated July 22, 1993.
2. GE Nuclear Energy, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (Non-Proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.
3. 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 (Non-Proprietary), March 1992; and Supplements 1 and 2.
4. GE Nuclear Energy, "Power Uprate Safety Analysis for NMP-2," Licensing Topical Report NEDC-31994P, Class III, Revision 1, (Proprietary), May 1993.

5. GE Nuclear Energy, "NMP-2 SAFER/ GESTR-LOCA, Loss-of-Coolant Analysis," Licensing Topical Report NEDC-31830P, Class III (Proprietary), Revision 1, November 1990.
6. NRC Safety Evaluation Report of Licensing Topical Report NEDC-31753P, dated March 8, 1993.
7. Letter to the NRC from B. R. Sylva, NMPC, response to request for additional information, dated October 6, 1994.

ATTACHMENT 2

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

FACILITY NAME NINE MILE POINT UNIT 2

SUMMARY OF REVIEW

POWER UPRATE REVIEW

NARRATIVE DISCUSSION OF LICENSEE PERFORMANCE- SAFETY ASSESSMENT/QUALITY VERIFICATION

The licensee submittal was complete, only a few questions in an RAI were needed. The RAI response was o.k. but, required a long time in submitting the information. This delayed the review and the requested completion date.

AUTHOR: R. FRAHM
DATE: 10/20/94