June 26, 1995

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

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 2 - CLARIFICATION AND CORRECTIONS TO SAFETY EVALUATION FOR LICENSE AMENDMENT NO. 66 (TAC NO. M87088)

Dear Mr. Sylvia:

The purpose of this letter is to transmit clarification and corrections to the safety evaluation (SE) which accompanied License Amendment No. 66 (power uprate). The revisions are minor and do not change the conclusions.

License Amendment No. 66 was issued April 28, 1995, authorizing an increase in the maximum power level of Unit 2 from 3323 megawatts thermal (MWt) to 3467 MWt. In your letter dated May 15, 1995, you identified a small number of minor inconsistencies in the accompanying SE, and in conversations with your staff, a few additional administrative errors in the SE were identified. Therefore, we are issuing corrected pages to the SE (Enclosure 1). A markup showing the revisions is provided as Enclosure 2. Any further questions regarding the SE should be directed to the undersigned at 301-415-1448.

Sincerely,

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. Corrected pages to SE
2. Markup to SE

cc w/encls: See next page

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

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REVISION PAGES FOR SAFETY EVALUATION

FOR NINE MILE POINT NUCLEAR STATION, UNIT 2

POWER UPRATE ADMENDMENT NO. 66

Remove Pages	Insert Pages
4	4
5	5
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24	24
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Enclosure 1

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 analytical limits for 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 psig. 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 113.9 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 psid as a result of power uprate will raise the pump suction pressure by approximately 15 psid, and the pump discharge pressure increases less (approximately 13 psid) than the suction pressure. 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 were evaluated by NMPC and are not affected by power uprate and ELLL. 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 will continue to provide calibration of flow control, loop flow and core flow instrumentation. As stated in NEDO-31897, tests should be performed to 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

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 113.9 Mlb/h, on LOCA results was evaluated at the 3536 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 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 the results 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 for core cooling. A coping evaluation was performed to demonstrate performance, based on the RCIC system. The coping time remains unchanged for power uprate. 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

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, 1995, to 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 will modify the preventative maintenance program to ensure 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 will modify 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 will be 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 Allowable Value 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 psig.
   Change Allowable Value from ≤ 129.6 psig to ≤ 136.4 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.

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's analyses were performed using methodology described in the UFSAR with the original licensing basis assumption at 3489 MWt (105% of current power level). The analyses indicate that the calculated offsite radiological consequences doses are within the dose acceptance criteria stated in 10 CFR Part 100 and also comply with the dose acceptance criteria 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 guidelines.

TABLE 1 - LOCA Radiological Consequences

	UFSAR 3489 MWt (rem)	SER 3489 MWt (rem)	Part 100 acceptance criteria
EAB:			
Whole Body Dose	6.3	2.6	25
Thyroid Dose	232.0	224.0	300
LPZ:			

On the basis of its review of NMPC's major assumptions, the methodology used in NMPC 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 and GDC 19 dose acceptance criteria, 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 negligible. 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 Control 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, 1971 Edition/Winter 1972 Addenda through 1974 Edition/Winter 1975 Addenda. 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 vessel high pressure scram analytical limit 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.

MARKUP OF SAFETY EVALUATION

FOR NINE MILE POINT NUCLEAR STATION, UNIT 2

POWER UPRATE AMENDMENT NO. 66

DOCKET NO. 50-410

Pages 4, 5, 9, 10, 23, 24, 25, 27, 29, 30

Enclosure 2

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

analytical limits for the

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).

psig

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.

increases less (approximately 13 psid) than the suction pressure.

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 115 113.9 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 psid as a result of power uprate will raise the pump suction pressure by 17 psig approximately 15 psid 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 were evaluated by NMPC and are not 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.~~ NMPC will continue to provide calibration of flow control, loop flow and core flow instrumentation. 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.

be performed to

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

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.

3536 113.9

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. the results X

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

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.

1995

3.13 Equipment Qualification (EQ)

NMPC's July 22, 1993, submittal was supplemented on April 10, 1994, to 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 ensure replacement of the affected components before the end of their qualified lifetimes.

will modify

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.
will modify

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.
will be

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.

Allowable Value

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 psig.
   Change Allowable Value from ≤ 129.6 psig to ≤ 135.4 psig. 6

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.

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's

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 The reconstituted analyses indicate that the calculated offsite radiological consequences doses are within the dose ~~reference values~~ stated in 10 CFR Part 100 and also comply with the dose ~~limits~~ acceptance criteria to control room operators given in General Design Criterion (GDC) 19.

X

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~~ ~~acceptance criteria~~ guidelines.

TABLE 1 - LOCA Radiological Consequences

X

	UFSAR 3489 MWt (rem)	SER 3489 MWt (rem)	~~Part 100 Limits~~ Part 100 Acceptance Criteria
EAB:			
Whole Body Dose	6.3	2.6	25
Thyroid Dose	232.0	224.0	300
LPZ:			

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 acceptance criteria ~~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 negligible. 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.

1971 Edition/Winter 1972 Addenda through
1974 Edition/Winter 1975 Addenda

3.17.2 Control 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.
vessel analytical limit

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