NIAGARA MOHAWK

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January 3, 1995
NMP2L 1518

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Nine Mile Point Unit 2
Docket No. 50-410
NPF-69

Gentlemen:

Subject: ***Proposed License Amendment - Uprated Operation, Response to Request for Additional Information***

In a letter to the Nuclear Regulatory Commission (NRC) dated July 22, 1993 (NMP2L 1397), Niagara Mohawk Power Corporation (NMPC) proposed a license amendment to allow Nine Mile Point Unit 2 (NMP2) to operate at an uprated power of 3467 megawatts thermal. During the course of the Staff's review of this proposed license amendment, the NRC has determined that additional information regarding compliance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME), as identified in its November 21, 1994 letter to NMPC, is required to complete its review of this matter. Attached to this letter are the Staff's questions and the requested additional information.

Niagara Mohawk has provided a copy of this response to the appropriate state representative.

Very truly yours,

C. D. Terry
Vice President - Nuclear Engineering

CDT/KWK/lmc
Attachment

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xc: Regional Administrator, Region I
Mr. B. S. Norris, Senior Resident Inspector
Mr. L. B. Marsh, Director, Project Directorate I-1, NRR
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ATTACHMENT

Note: Unless otherwise stated, section numbers apply to Enclosure 3 of the proposed power uprate license amendment provided by Niagara Mohawk in the letter dated July 22, 1993 (NMP2L 1397).

QUESTION 1

In Section 2.5.1, the evaluation did not address the effects of the power uprate, such as the increase of the bottom head pressure and the high pressure scram setpoint, on the structural and functional integrity of the control rod drive system (CRDS). State the basis for determining the acceptability of the CRDS regarding compliance with the American Society Mechanical Engineers Boiler and Pressure Vessel Code (ASME) Code. The information should include the ASME Code edition that was used in the evaluation, corresponding allowables, the calculated maximum stresses, deformation, and fatigue usage factor for the uprated conditions, and assumptions used in the calculations.

RESPONSE

The Nine Mile Point 2 (NMP2) CRDS was evaluated consistent with a nominal 1020 psig reactor dome pressure and an additional 35 psid for the vessel bottom head. The basis for determining the acceptability of the CRDS regarding compliance to the ASME Code is addressed in the response to question 4 below. However, the acceptability of the control rod drive mechanism (CRDM) is discussed in the following paragraphs of this response to question 1.

The CRDM structural and functional integrity was deemed acceptable for the vessel bottom head pressure of 1055 psig. The components of the CRDM designated as a primary pressure boundary have been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The NMP2 CRDMs were supplied and evaluated to the 1971 Edition/Winter 1972 Addenda through 1974 Edition/Winter 1975 Addenda.

The limiting component of the CRDM is the indicator tube which has a calculated stress of 20,795 psi versus an allowable of 31,050 psi (emergency condition). The maximum stress is due to a maximum CRDM internal hydraulic pressure of 1750 psig. When considering "New Loads" (hydrodynamic loads included in the original licensing basis), the limiting component of the CRDM continues to be the indicator tube which has a maximum calculated stress of 37,600 psi versus an allowable of 40,000 psi (faulted condition).

The analysis for cyclic operation of the CRDM was conservatively evaluated in accordance with ASME Code, NB-3222.4. All requirements of NB-3222.4 are satisfied even when considering the increased power uprate vessel bottom head pressure, thereby satisfying the peak stress intensity limits governed by fatigue. It should be noted that the CRDM has been successfully tested for all operational modes at simulated reactor vessel pressures up to 1250 psig saturated conditions, which bounds the uprated high pressure scram analytical setpoint pressure of 1086 psig. Additional analysis shows that the maximum usage factor calculated based on NB-3222.4(e) is 0.15 for the CRDM main flange, which is less than the allowable limit of 1.

As indicated above, successful testing has been performed at 1250 psig and the calculated maximum stress levels are below the allowables. Therefore, NMPC has concluded that the deformations are not a concern at power uprate conditions.

The CRDS is capable of providing 250 psid between the hydraulic control unit and the reactor vessel for control rod insert and withdraw operation. At power operation, the primary scram pressure is provided by the reactor vessel pressure. Therefore, the CRDS will perform all of its safety function operations at NMP2 power uprate conditions.

QUESTION 2

In Section 3.3.2, the evaluation did not address the ASME Code edition used for evaluating stresses and allowables for the reactor vessel and internals. Please provide this information and list the maximum stresses, fatigue usage factor, and locations of highest stressed areas for both the current design and the uprated power conditions.

RESPONSE

The 1971 Code (ASME, "Rules for Construction of Nuclear Vessels," ASME Boiler and Pressure Vessel Code Section III, 1971 Edition with Addenda to and including Winter 1972), which is the code of construction, is the governing code used in evaluating stresses and allowables for the reactor pressure vessel (RPV) and internals.

The maximum stresses, fatigue usage factors, and locations of highest stressed areas are given in Section 2.0 of NEDC-32015, Revision 1, entitled "Nine Mile Point 2 Fatigue Evaluation Power Uprate Operating Conditions." This NEDC, which is provided at the end of this attachment, provides the basis for concluding that the fatigue evaluation of the highest stressed areas satisfies the requirements of the above ASME Code edition.

QUESTION 3

In Section 3.3.3, the calculated fatigue usage factor provided in this section for the feedwater nozzle is nearly 1.0 for the uprated power conditions. Provide a detailed discussion on the analysis methodology, assumptions, the basis for acceptability of stress level and fatigue value in relation to allowable specified in the ASME Code edition of record.

RESPONSE

A detailed discussion of the analysis methodology, assumptions, the basis for acceptability of stress level and fatigue values in relation to the allowable specified in the ASME Code edition of record is provided in Section 2.0 of NEDC-32015.

In summary, the stresses have not changed significantly due to the uprate condition. The calculated usage factors of 0.916 at the critical stainless steel clad location and 0.965 for the carbon steel location are actually below the respective values of 0.9503 and 0.968 in our current design base analysis. The decrease was a result of providing more accurate values for allowable stress cycles (previously estimated allowable stress cycles were unnecessarily conservative), which in turn decreased the usage factor and more than offset the small increase in alternating stresses.

QUESTION 4

In Section 3.5, provide an evaluation of the increased safety/relief valve (SRV) discharge dynamic loads and the increased main steam isolation valve closure dynamic loads on the main steam and SRV discharge piping and supports. Also specify the ASME Code edition used for evaluation of the nuclear steam supplier piping and pipe supports and provide maximum piping stresses and fatigue usage factors at critical locations.

RESPONSE

The design SRV dynamic loads for the current licensing basis are based upon a setpoint pressure that exceeds the highest SRV safety set point pressure for power uprate. Therefore, for power uprate, there is no increase in the calculated SRV discharge dynamic loads since the highest SRV setpoint was not increased. The SRV opening duration during transient events such as MSIV closure, may increase slightly due to power uprate. However, at high vessel pressure, the condensation related load due to steady steam flow is negligible and is unaffected by the duration.

The evaluation of the main steam piping included an assessment of the increase to the turbine stop valve (TSV) closure load. Since this load is significantly more severe than the MSIV closure load (due to differences in closure times), it is not necessary to assess the MSIV load since it is enveloped by the TSV load.

On December 21, 1994, NMPC discussed the second sentence of question 4 with the Staff. Specifically, NMPC proposed providing the following alternative information to satisfy the Staff's request for additional information regarding the ASME Code Edition, maximum piping stresses and fatigue usage factors at critical locations.

The power uprate evaluation of piping and pipe supports was performed using the original Code Edition (including code year and addendum) which is the licensing basis of the plant. In general, the original Code Edition for piping classes 1, 2 and 3 is the ASME Boiler and Pressure Vessel Code Section III, Division I, Nuclear Power Plant Components Dated July 1974. The evaluation of piping and pipe supports, in all cases, demonstrates that the maximum stress levels and fatigue usage factors for the power uprate condition satisfy the original specific Code requirements for stress allowables and fatigue usage factors. Therefore, compliance to the original specific ASME Code Edition, defined as the Code of Record, is maintained for the power uprate condition for plant systems, including the following systems identified in Section 3.5:

a. Main Steam (including RPV bottom head drain and head vent lines) and SRV discharge piping,

b. Recirculation,

c. Feedwater,

d. Control Rod Drive,

e. Reactor Core Isolation Cooling,

f. Residual Heat Removal,

g. High Pressure Core Spray,

h. Low Pressure Core Spray,

i. Reactor Water Cleanup and

j. Standby Liquid Control

QUESTION 5

In Section 3.5, it is stated that the evaluation of the Class 4 feedwater piping is still in progress and that it will be completed and resolved before implementation of power uprate. State how this piping is evaluated for the power uprate. Specify the analysis methodology, assumptions and code of record.

RESPONSE

Class 4 feedwater piping was originally stress analyzed (design basis) using ANSI B31.1, Power Piping Code, 1973 Edition including Winter 1973 Addenda, dated December 31, 1973. Stress analysis was performed for dead weight, thermal (considering various operating temperature conditions), thermal stratifications, and fluid transients. Pressure stress computation was based on the design pressure.

This piping has been evaluated for power uprate conditions as follows:

a. Operating pressures: The original qualification of the piping used design pressures in the Code equations. Since the operating pressures for power uprate conditions are bounded by original design pressures, the changes in operating pressures do not increase the calculated pressure stresses.

b. Operating Temperature: Operating temperatures utilized in the original analysis envelope those for power uprate. Therefore, there is no increase in the calculated stresses.

c. Fluid transients: Fluid transient loads for power uprate conditions have been generated and compared with those in original design. Due to substantial conservatism in the original analysis, original design loads are found to be enveloping. Therefore, there is no increase in the calculated stresses from fluid transient loads.

For power uprate, pipe stresses are qualified to meet ANSI B31.1, Power Piping Code, 1973 including Winter 1973 Addenda dated December 31, 1973. Pipe supports are qualified using AISC and AWS allowables for ANSI B31.1 loading conditions.

QUESTION 6

In Section 3.12, identify the critical balance of plant (BOP) piping systems and components evaluated for the power uprate, but were not included in Section 3.5 (i.e., SRV discharge line, main steam drain). Specify the ASME Code edition used in the power uprate evaluation of limiting BOP piping systems and pipe supports including anchorages. Provide a discussion on the acceptability of auxiliary systems such as safety related heat exchangers and chilled water systems, with respect to stress levels (including fatigue consideration) for operation at uprated power conditions.

RESPONSE

In addition to the systems listed under Section 3.5 of the submittal, piping for the following BOP systems have been evaluated for power uprate.

1. Condensate
2. Reactor Vessel Instrumentation
3. Moisture Separator Reheater Vents and Drains
4. Circulating Water
5. Off Gas
6. Auxiliary Condensate
7. Auxiliary Steam
8. Turbine Plant Miscellaneous Drains
9. Turbine Generator Gland Seal and Exhaust Steam
10. Extraction Steam
11. HP Feedwater Heater Drains
12. LP Feedwater Heater Drains
13. Moisture Separator Vents and Drains
14. Feedwater Pump Recirculation

The above systems, their components and supports have been evaluated at the increased pressure, temperature and flow conditions corresponding to power uprate. The evaluation of the above piping systems and supports used the following codes:

- Piping classes 2 and 3 used ASME Boiler and Pressure Vessel Code Section III, Division I, Nuclear Plant Components dated July 1974.

- Piping class 4 used ANSI B31.1, Power Piping Code, 1973 including Winter 1973 Addenda, dated December 31, 1973.

- Pipe Supports used American Institute Steel Construction (AISC) and American Welding Society (AWS).

The evaluations of the above systems at the power uprate conditions demonstrate continued compliance to the above codes.

The residual heat removal (Section 3.9) and the spent fuel (Section 6.3) heat exchangers' stress levels are insignificantly affected by the power uprate of NMP2. For power uprate, the post-LOCA operating pressure and temperature of the process water side for the residual heat removal heat exchangers is within the original design requirements. The shutdown cooling mode operating conditions of the residual heat removal heat exchangers are not affected by power uprate. The spent fuel pool heat exchanger operational heat load increases slightly for the power uprate condition; however, the as-built design of this heat exchanger envelopes the power uprate condition. Therefore, the calculated stress levels of these heat exchangers do not increase at the power uprate condition.

Some safety related and non-safety related unit coolers are affected by the power uprate due to the increase in the operational heat loads in some plant areas. However, these changes in operational heat loads are bounded by the original design.

The chilled water systems (control building chillers and ventilation chilled water) are not affected by the power uprate.

The remaining critical BOP components were evaluated for the power uprate condition using the ASME Code of Record. This evaluation demonstrates that the power uprate stress levels and usage factors meet the original ASME Code Criteria.

Upstream of the inboard main steam isolation valves, main steam drain piping is ASME Code Class 1. As presented in Section 3.5 of the submittal, the effects (pipe stresses and cumulative usage factors) due to increased pressure, temperature and flow have been evaluated by examining the margin between the calculated stress level and the code allowable for the original design. Pipe supports and penetrations were evaluated either by examination of the remaining margin or by recalculation of the maximum stress levels. These power uprate evaluations maintain compliance with the original code of record.

Between the inboard and outboard main steam isolation valves, the main steam drain piping is ASME Code Class 2. Operating temperature and pressure values used in the original analysis of this piping envelope those for power uprate. Since this piping does not experience any significant fluid transient load, the existing analysis envelopes the power uprate condition.

Downstream of outboard isolation valve, main steam drain piping is designed to ANSI B31.1 Power Piping Code and is subject to slightly higher temperature (5 F) due to power uprate. The small increase in temperature for this piping has been evaluated and found to have an insignificant impact on pipe stresses, equipment nozzles and pipe supports due to conservatism in the original design.

A discussion of the evaluation of the SRV discharge line is provided in the response to question 4.

QUESTION 7

In Section 10.5, a recent abnormal reactor recirculation pump vibration issue has been reported by the Susquehanna licensee during testing for the power uprate conditions. Provide an evaluation of the increased flow-induced dynamic loads on the recirculation piping and components such as pumps and flow control valves, to assure that excessive recirculation pump vibration will not occur at NMP2 for the uprated power conditions.

RESPONSE

As stated in Sections 3.5 and 10.3.3, the piping and mechanical components were found to have adequate capability for uprated operating conditions. These evaluations included consideration of the full recirculation and core flow range planned for uprated operation (up to 113.9 Mlbm/hr, 105% of rated core flow).

There are significant differences between NMP2 and the Susquehanna experience. They are:

a. NMP2 is already licensed at the 105% core flow condition and has accumulated operating experience at core flows up to approximately 104.7% of rated core flow (i.e., significant operating experience in the lower half and limited operating experience in the upper half of the increased core flow range) while Susquehanna had not done so before the power uprate and increased core flow testing.

b. NMP2 is a recirculation flow control valve plant with two-speed recirculation pump motors (high speed only at full power and flow) whereas Susquehanna is a variable pump speed unit.

c. When NMP2 extends operation into the increased core flow region, the pumps remain at near synchronous speed and the flow control valves are opened further (smaller pressure drop) whereas Susquehanna increases the speed of the pumps into a range not used during operation below the increased core flow range.

NMP2 (like Susquehanna) does require a very small increase (less than 0.5%) in recirculation drive loop flow to achieve the same core flow at uprated power versus the original rated power level (because of slightly increased core pressure drop). The increase is achieved by a slight opening of the flow control valve (smaller pressure drop and reduced valve loading). The pump remains at the same speed. The effects of the small flow change are well within the acceptable loads and capability of the recirculation system and associated equipment.

In summary, there are no reasons to expect any increase in recirculation system vibration for NMP2 uprated power operation because NMP2 is not increasing the range of core flow coincident with power uprate and the unit has already operated at the planned recirculation conditions without difficulty.