March 12, 1984

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MAY

Docket No.: 50-410

Mr. Gerald K. Rhode
Senior Vice President
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Rhode:

Subject: SER Review Meeting for Mechanical Engineering for Nine Mile Point, Unit 2 (NMP-2)

The Mechanical Engineering SER review of Nine Mile Point 2 is tentatively scheduled for the week of June 4, 1984. Enclosure 1 contains a list of issues and NRC staff positions to be discussed during the SER review. This information was provided to Norm Rademacher of your staff on February 24, 1984. An agenda for this meeting should be prepared by your staff based on the issues listed in Enclosure 1. These issues are considered open at this time.

We anticipate this meeting will be held over a 3-5 day period at the SWEC offices in Cherry Hill, New Jersey. At this extended meeting, it is expected that most of the open items noted in the enclosure will be resolved. Therefore, any GE, SWEC and NMPC representatives necessary to both discuss technical issues and make binding commitments should be present at this meeting.

Any questions concerning this meeting and the enclosed information should be addressed to the licensing project manager, Mary Haughey, at (301) 492-7897.

Sincerely,

Original signed by:

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc w/ enclosure:
See next page

M F Haughey

DL:LB#2/PM	DL:LB#2/BC
MFHaughey:pt	ASchwencer
3/8/84	3/12/84

Nine Mile Point 2

Mr. Gerald K. Rhode
Senior Vice President
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

cc: Mr. Troy B. Conner, Jr., Esq.
Conner & Wetterhahn
Suite 1050
1747 Pennsylvania Avenue, N.W.
Washington, D. C. 20006

Mr. Richard Goldsmith
Syracuse University
College of Law
E. I. White Hall Campus
Syracuse, New York 13210

Mr. Jay Dunkleberger, Director
Technological Development Programs
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Ezra I. Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Resident Inspector
Nine Mile Point Nuclear Power Station
P. O. Box 99
Lycoming, New York 13093

Mr. John W. Keib, Esq.
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Jay M. Gutierrez, Esq.
U. S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

NINE MILE 2 SER QUESTIONS

SECTION 3.6.2

210.17 Branch Technical Position MEB 3-1 requires that certain stress and cumulative usage factor limits be met in the break exclusion zone. The criteria contained in the FSAR are not in compliance with these limits. Provide justification for the criteria used. In particular, address those cases for Equation (10) exceeding 2.4 S_m and the cumulative usage factor exceeding 0.1 for Class 1 piping.

210.18 Have occasional loads been considered in the evaluation of the sum of Equations (9) and (10) when comparing to the limits for Class 2 piping in the break exclusion area?

210.19 Provide assurance that 100% volumetric inservice examination of all pipe welds in the break exclusion area will be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

210.20 Breaks in non-nuclear high energy piping not seismically analyzed (nor qualified) should be postulated at those locations which produce the greatest effect on an essential component or structure irrespective of the fact that the high stress or fitting criteria might not require a break to be postulated. Provide assurance that the above criteria have been met.

210.21 What criteria are used for postulating moderate energy leakage cracks inside containment?

210.22 Discuss how high energy leakage cracks are considered.

210.23 Discuss how pipe whip and jet impingement effects were determined for those postulated breaks in high energy piping that is not restrained.

210.24 Provide assurance that the tip deflection of a restrained whipping pipe does not adversely affect nearby safety-related components from performing their safety-related function.

210.25 Describe in more detail the design procedures and methodologies used in the jet impingement analyses. Specifically, address 1) the jet loads and jet configurations used for circumferential and longitudinal breaks, 2) how targets are determined, and 3) the acceptance criteria used to evaluate the effects on safety-related components and structures.

210.26 Provide the criteria used in the design of pipe rupture restraints including the auxiliary steel used to support the pipe rupture restraint. Provide assurance that the pipe rupture restraint and supporting structure cannot fail during a seismic event.

210.27 Provide the design criteria for pipe rupture restraints that also support piping.

210.28 In order to assure the pipe break criteria has been properly implemented, the Standard Review Plan requires the review of sketches showing the postulated rupture locations and summaries of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. The vast majority of this information in the FSAR is either preliminary or incomplete. Please provide a schedule for the completion of Tables 3.6A-2 through 3.6A-60 and Figures 3.6A-12 through 3.6A-39.

210.29 On page 3.6A-28 an amplification factor of between 1.0 and 1.1 to

account for pipe rebound is discussed. Provide justification for the use of an amplification factor of less than 1.1.

210.30 Provide justification for shape factors of less than unity as discussed on page 3.6A-35 of the FSAR.

210.31 Appendix 3C is very incomplete. Provide a schedule for its completion.

210.32 Provide a list of all instances where full break area opening times in excess of one millisecond were used. See page 3.6B-8.

210.33 Provide justification for using a thrust coefficient of less than 1.26 for saturated steam and 2.0 for subcooled water as discussed on page 3.6B-9.

210.34 Provide a list of all instances where mechanistic approaches were used to reduce break areas as discussed on page 3.6B-6.

210.35 Provide the basis for assuring that the feedwater isolation check valves can perform their function following a postulated break of the feedwater line outside containment.

210.36 Discuss the types of protection used to mitigate the effects of jet impingement on safety-related components and structures.

SECTION 3.9.2

210.37 Provide the acceptance criteria to be used in determining if the vibration loads observed or measured during the pre-operational

testing are acceptable. Specifically, address how the vibration amplitudes will be related to a stress load and what stress levels will be used for both steady-state and transient vibration.

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

The essential instrumentation lines to be inspected should include (but are not limited to) the following:

a. Reactor pressure vessel level indicator instrumentation lines (Used for monitoring both steam and water levels).

b. Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow).

c. Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation).

d. Control rod drive lines inside containment (not normally pressurized but required for scram).

210.38 Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety-related systems and

components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in Tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9. This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a minimum verify the following:

1. There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.
2. The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.
3. Snubbers are not seized, frozen or jammed.
4. Adequate swing clearance is provided to allow snubber movement.
5. If applicable, fluid is to be recommended level and is not leaking from the snubber system.
6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250°F should be verified as follows:

a. During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

b. For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

c. Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepancies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

210.39 Please provide a statement as to the compliance with NUREG-0619, "BWR Feedwater Nozzles and Control Rod Drive Return Line Nozzle Cracking".

210.40 Provide the basis used for the design of piping supports and anchors which separate seismically designed piping and non-seismic Category I piping. Include in your discussion the loads and load combinations used and how the local pipe wall stresses are considered.

210.41 Describe the design considerations given to assure that an

adequate number of modes have been used in the dynamic piping anlayses performed for:

1) seismic loadings,
2) SRV loadings,
3) LOCA loadings, and
4) hydraulic transients (e.g. steam and water-hammer).

210.42 Explain how in the design process the reinforcement thickness of branch connections are determined for both internal pressure and mechanical loads and incorporated into the fabricated piping. Provide assurance that all branch connections decoupled from the main run piping on the piping analytical model are designed and fabricated to the required reinforcement area.

210.43 The staff finds insufficient information describing the design of safety-related HVAC ductwork and supports. Provide the design basis used for qualifying the HVAC ductwork and support structural integrity.

SECTION 3.9.3

210.44 Provide the basis for assuring that ASME Code Class 1, 2, and 3 piping systems are capable of performing their safety function under all plant conditions. Describe the methodology used to assure the

functional capability of essential piping system when service limits C or D are specified.

210.45 Provide a discussion of the design considerations used for safety and relief valve loads and piping reactions. Include in your discussion the basis for assuring that the valve end loads and the support arrangement for the affected piping are acceptable.

210.46 Describe those short-term and long-term actions being taken to preclude the occurrence of cracking in jet pump hold down beams as described in IE Bulletin 80-07.

210.47 Describe briefly the design considerations given to the piping stress analyses for the mainsteam piping and attached safety relief valve discahrge piping for the alternate shutdown cooling mode. Specifically address the capability of the spring hangers to accommodate the additional weight of water during this mode.

210.48 Provide assurance (or a commitment) that the design of all safety-related mechanical components and their supports can withstand the effects of safety-relief valve discharge laods as defined in NUREG-0802, "Safety/Relief Valve Quencher Loads = Evaluation for BWR Mark II and III Containments."

210.49 Provide assurance (or a commitment) that the design of all safety-related mechanical components and their supports can withstand the effects of loss-of-coolant accident loads as defined in NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria."

210.50 Provide the basis for assuring that a fatigue crack will not occur 1) in the safety relief valve discharge piping in the suppression pool wetwell airspace and 2) in the suppression pool downcomers. The staff requests that ASME Code Class 1 piping fatigue evaluation

be performed and should include all cyclic loadings due to normal operation, seismic, SRV discharge, LOCA chugging, and condensation oscillation loads (as appropriate).

210.51 Porvide for the staff review, typical examples of the amplified building response spectra used in the design of piping systems and including the following loadings:

a) seismic OBE
b) seismic SSE
c) SRV loads
d) LOCA-related loads

210.52 Briefly, describe the attenuation of the hydrodynamic loads in the plant and describe to what extent safety-related components are designed to these loadings in the various areas in the plant.

210.53 Describe the design considerations given to the piping in the suppression pool wetwell with respect to stability of the piping and its supports during a LOCA pool swell event.

210.54 Using the guidance of NUREG-0609, provide the methodology used and the results of the annulus pressurization (AP) analysis (asymmetric LOCA loads) for the reactor system and affected components including the following:

1. reactor pressure vessel and supports,
2. core supports and other reactor internals,
3. control rod drives,
4. ECCS piping attached to the reactor coolant system,
5. primary coolant piping, and
6. piping supports for affected piping systems.

The results of the above analysis should specifically address the effects of the combined loadings due to annulus pressurization and an SSE.

210.55 The staff review finds insufficient information regarding the design of component supports. Per SRP Section 3.9.3, our review includes an assessment of design and structural integrity of the supports. The review addresses three types of supports: (1) plate and shell, (3) linear, and (3) component standard types. For each of the above three types of supports, provide the following information (as applicable) for our review:

(a) Describe (for typical support details) which part of the support is designed and constructed as component supports and which part is designed and constructed as building steel (NF vs AISC jurisdictional boundaries).

(b) Provide the complete basis used for the design and construction of both the component support and the building steel up to the building structure. Include the applicable codes and standards used in the design, procurement, installation, examination, and inspection.

(c) Provide the loads, load combinations and stress limits used for the component support up to the building structure.

(d) Provide the deformation limits used for the component support.

(e) Describe the buckling criteria used for the design of component support.

210.56 The staff's review of your component support design finds that additional information is required regarding the design basis used for bolts.

(a) Describe the allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections.

(b) Provide a discussion of the design methods used for expansion anchor bolts used in component supports.

(c) Identify where in the plant high strength bolts have been used.

210.57 It is the staff's position that for the design of component supports, stresses produced by seismic anchor point motion of piping and the thermal expansion of piping should be categorized as primary stresses. Confirm that Nine Mile Point 2 meets this criteria.

210.58 Provide a discussion of the use of stiff pipe clamps as addressed in IE Information Notice 83-80.

210.59 Provide assurance that any snubbers used as a vibration arrestor has properly considered the cyclic loadings which might cause fatigue failure.

210.60 Valve discs are considered part of the pressure boundary and as such should have allowable stress limits. Provide these limits for our review.

210.61 Provide the stress categories and limits for core support/structures and include the applicable codes used for evaluation of the faulted condition.

<u>SECTION 3.9.6</u>

210.62 There are several safety systems connected to the reactor coolant

pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements which will state the acceptable leak rate testing frequency shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute (GPM)

for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

210.63 Provide a schedule for completion of your program for inservice testing of pumps and valves including any request relief from ASME Section XI requirements.