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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: Nine Mile Point 2 OL Safety Review - Request for Additional Information

Enclosures 1 through 8 of this letter identify additional information required for our review of the safety aspects of your application for an operating license for Nine Mile Point 2 in the following areas:

	SRP Section	Enclosure
Structural Engineering	3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1-3.7.4, 3.8.1-3.8.5	
Geotechnical Engineering	2.5.4-2.5.6	2
Materials Engineering	4.5.2, 5.2	3
Auxiliary Systems	see questions	4
Reactor Systems	15.0 (Chapter 15 - Accident Analyses)	5
Accident Evaluation	6.5.3, 6.7, 15.6.5	6
Radiologic Assessment	12.1 - 12.5	7
Core Performance - Thermal Hydraulics	4.4	8

OFFICE							
SURNAME							
DATE							

As discussed in our letters of July 25, 1983, and August 12, 1983, additional requests for information will be transmitted to you as we complete our reviews of the remaining sections.

Consistent with the licensing review schedule for Nine Mile Point Unit 2, responses to these requests for additional information should be submitted as changes to the FSAR by October 27, 1983.

If you have any questions concerning the enclosed requests for additional information, please call the licensing project manager, Mary F. Haughey at (301) 492-7897.

Sincerely,

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

Enclosures:
As stated

cc: See next page

M.F. Haughey A. Schwencer

OFFICE	DL:LB#2/PM	DL:LB#2/BC					
SURNAME	MHaughey:pt	ASchwencer					
DATE	8/15/83	8/15/83					

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 126
Lycoming, New York 13093

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

NINE MILE POINT 2 - OL REVIEW
STRUCTURAL QUESTIONS

220.3 SRP. 3.3.2-II FSAR 3.3.2.2.4 3.8.4.3	With regard to tornado load combinations identify the controlling load combinations used for design of structures or structural elements. Provide example of design calculations covering the controlling load combination.
220.4 SRP 3.4..2-II FSAR 3.4.1.1.3	The FSAR stated that all construction joints below el. 261 ft. in the foundation mats and exterior structural walls of Category I structures have water stops or other means to prevent leakage through the joints. What are the so-called "other means"? Identify where these were used and the basis for their selection instead of water stops. Are the water stop materials properly selected to resist deteriorations due to potential environmental effects such as aging, heating, radiation, and chemicals? Provide details of the materials used, their expected service environment, and their expected performance under the influence of the same. Also provide pertinent test data demonstrating their expected level of performance.
220.5 SRP 3.5.3-II FSAR 3.5.3	Provide a comparison of tornado missile barrier thickness used for all Category I concrete structures at the plant and those listed in NRC NUREG 0800, SRP Section 3.5.3 Table 1, Revision 1, dated July 1981. Are there openings in the walls or roofs of Category I structures which would allow a tornado missile to pass through? If so, what measures were taken to protect safety related systems and components which may be located in the vicinity of missile paths.
220.6 SRP 3.7.1 -II FSAR 3.7.1.2A	Indicate whether for Nine Mile Point 2 plant seismic design, the ratios of vertical design response spectral values to the horizontal design response spectral values comply with the position of Regulatory Guide 1.60, i.e., the ratio varies for different frequencies. If not, assess the impact on seismic design and justify the deviations.
220.7 SRP 3.7.2-II FSAR 3.7.2.1A	With respect to seismic analysis method, your discussion of the method to account for the effects of maximum relative displacements among supports of Category I structures, systems and components are not clear. Provide a detailed discussion of how the effects of relative displacements were accounted for and the basis thereof.

220.8 SRP 3.7.2-II FSAR 3.7.2.8A 3.8.4.1 3.8.5.1	Provide a tabulation of the "structural gaps" provided for Category I structures along with an adjacent tabular listing of the worst computed gaps between structures. Discuss the basis for the selection of a minimum of six inches structural gap. Also demonstrate that adequate separations between Category I structures has been provided.
220.9 SRP 3.7.3-II FSAR 3.7.3.5A	With respect to static analysis method, provide justification for applying a static coefficient 1.3 to the peak acceleration rather than the staff accepted value of 1.5.
220.10 SRP.3.7.3-II FSAR 3.7.3.12A	Describe in detail the methods used for seismic design and analysis of Category I tunnels. Also provide a description of pertinent design criteria and results of design/analysis used for the buried Category I tunnels.
220.11 SRP 3.7.4-II FSAR 3.7.4A	Provide details of a seismic instrumentation inservice surveillance program. The staff's position is outlined in NUREG 0800, SRP Section 3.7.4-II.5.
220.12 SRP 3.8.1-II FSAR 3.8.1.5 3.8.3.5	With respect to the allowable stress for tangential shear in the concrete, SRP Section 3.8.1-II.5 stated that under no conditions shall the tangential shear carried by the concrete exceed 40 psi and 60 psi for the load combinations representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively. The FSAR stated that tangential shear stress carried by concrete were: For $0.01 \lesssim 0.01$ $\quad Vc = 12{,}000\ p$ For $0.01 \lesssim p \lesssim 0.025$ $\quad Vc = 93 + 2700\ p$ This is a deviation from the SRP position. Provide the results of actually computed stresses and justify the deviation.
220.13 SRP 3.8.1-II FSAR 3.8.1.3.2	Indicate whether the design and analysis of the liner plate and its anchorage system are in conformance with the requirements of ASME Code Section III, Division 2, Subsection CC-3600 and CC-3700. If not, identify and justify the deviations.

220.14 SRP 3.8.1-II FSAR 3.8.1.4 6.A.2.4.2	The FSAR stated that for the determination of the dynamic response of the containment structure subject to SRV discharge and LOCA loads, a finite element based computer program, i.e., dynamic stress analysis of axisymmetric structures under arbitrary loading, developed by S. Ghosh and E. Wilson and modified by SWEC was utilized. Provide discussion on the nature and extent of the code modification implemented and the latest code validation information.
220.15 SRP 3.8.I-II FSAR 6A.2.2	For combining various dynamic loads, such as LOCA, SRV, and OBE/SSE to determine the response of containment structure, it is the NRC staff's position that the absolute sum method should be used unless actual time histories of dynamic load occurrences are combined. It is not clear based on the FSAR whether your load combination procedure complied with the staff position or not. If not, please justify the deviations.
220.16 SRP 3.8.4-II FSAR 3.8.4.1.5 3.8.4.1.6	Data on FSAR Table 6A.5-1 to 6A.5-11 and Table 6A.6.1 to 6A.6.2 are not completed. Provide available analysis results and complete the contents of the tables for review by the staff.
220.17 SRP 3.8.4-II FSAR 3.8.4.1.5 3.8.4.1.6	Provide more detail drawings including sections, elevations and connections of Category I intake (discharge) tunnels. Are the tunnels lined with reinforced concrete?
220.18 SRP 3.8.4-II FSAR 3.8.4.4	Do you have any masonry walls at Nine Mile Point 2 plant, the failure of which could damage or affect the functioning of safety related structures, systems and components? If so, please demonstrate that the design, analysis and construction of these walls comply fully with the staff's acceptance criteria for masonry walls provided in Appendix A to SRP Section 3.8.4, NUREG-0800.
220.19 SRP 3.8.4-II FSAR 3.8.4.6.2	With respect to mechanical rebar splicing systems, identify structures or structural components that: (1) Dywidag threadbar system splices are used. (2) Welding of reinforcing steels are used. (3) Locations and bar sizes used for both (1) and (2). (4) Discuss the mechanical splice criteria used with respect to compliance with the requirements of Regulatory Guide 1.136.
220.20 SRP 3.8.5-II FSAR 3.8.5.1.1	(1) With respect to the design and analysis of reactor building mat, provide results of your analysis such as design moments and shears for the foundation mat at various critical sections. Also provide a detailed discussion of how these moments and shears are accommodated in the design.

(2) Demonstrate that provisions of the applicable code are fully met in your design of key foundation mat sections.

220.21 SRP 3.8.5-II FSAR 3.8.5.5	Provide details of stability analyses of Category I structures and demonstrate that the factors of safety against floating, sliding and overturning, as shown in NUREG-0800 SRP Section 3.8.5-II.5, are met.
220.22 SRP 3.8.4-II FSAR 9.1.2.2	Provide the load combinations, analysis procedures and acceptance criteria used in the design of fuel pool liner and slab. Indicate how the leak tight integrity of the fuel pool liner and structural integrity of the pool slab will be maintained in the event of a heavy drop accident.
220.23 SRP 3.8.4-II FSAR 9.1.2.2	Provide the sketches of the mathematical models used in the design of spent fuel racks. Describe in detail, the methods of analysis, including treatment of non-linear conditions due to gaps or friction, friction forces, boundary conditions, spring-mass locations, fluid modeling and damping considerations. Describe the methods by which seismic and other loads are applied to the racks and the pool.
220.24 SRP 3.8.4-II FSAR 9.1.2.3	Indicate whether material, fabrication, welding, and quality control of the spent fuel racks are in conformance with applicable provisions of Subsection NF of the ASME code. If not, identify and justify the deviations.
220.25 SRP 3.8.1-II FSAR 6A.5.1	With regard to the structural model used for hydrodynamic load analysis, provide description on fluid modeling. Also, provide the assessment of interactive effect between the fluid mass and the structures on the structural response to SRV loads.
220.26 SRP 3.8.1-II FSAR 6A 7.2.1.2	With regard to the design and analysis of containment basemat liner, the FSAR stated that a fatigue analysis of the basemat liner will be performed. When will this analysis be completed? Provide the report of the analysis for review by the staff.

220.27 SRP 3.5.3-II.1	The design of Category I structures and barriers to withstand the effects of missile impact on Nine Mile Point 2 is based on missile spectrum A of SRP Section 3.5.1.4, revision 2 instead of those shown on the Table 2 of SRP Section 3.5.3, Revision 1 (July 1981). Please justify the deviation in the missle spectrum selection and assess its potential impact on the structural barrier design.
220.28 SRP 3.7.2-II.11	To account for accidental torsion, the SRP states that an additional eccentricity of ± 5% of the maximum building dimension at the level under consideration should be assumed in the seismic analysis of Category I structures. For Nine Mile Point 2, however, the accidental torsion effects for the additional ± 5% were not considered in the analysis. Provide an assessment of the adequacy of the analysis considering the effects of accidental torsion.
220.29 SRP 3.8.1- II.4.j	Provide the ultimate capacity analysis of the containment for Nine Mile Point 2.
220.30 SRP 3.8.1- II.4.1	A concrete containment design report should be prepared and made available for review during the structural design audit to be performed by the staff at a later date. A suggested format is included in Appendix C to SRP Section 3.8.4, but as long as substantial structural design information is included in its content, some deviation from that format will be acceptable.
220.31 SRP 3.8.3- II.2	The SRP specifies that interior structures of containment should be designed in accordance with the requirements of ACI 349 code as augmented by Regulatory Guide 1.142. The Nine Mile Point 2 interior structures are designed in accordance with the requirements of ACI 318-77 code. Identify and justify all deviations of the interior structural design from the applicable requirements of the ACI 349 code as amended by Regulatory Guide 1.142.
220.32 SRP 3.8.3- II.4.e SRP 3.8.4- II.4.d SRP 3.8.5 II.4.e	Provide design reports for future structural design audit work covering SRP Sections 3.8.3, 3.8.4 and 3.8.5. A suggested format is shown on Appendix C to SRP Section 3.8.4. As long as the design reports provide sufficient structural design information, some deviation from that format is acceptable.

220.33 SRP 3.8.4-II-2 SRP 3.8.5-II.2	The SRP states that Category I structures should be designed in accordance with ACI 349 code as augmented by Regulatory Guide 1.142. The Nine Mile Point 2 Category I structures are designed in accordance with the requirements of the ACI 318-77 code. Identify and justify all deviations of the Category I structural design from the applicable requirements of ACI 349 code as amended by Regulatory Guide 1.142.

Nine Mile Point, Unit 2

Review Questions - Geotechnical Engineering

241.2 (SRP 2.5.4.1)	State the range and mean value of the core recovery and rock quality designation (RQD) ratios for each significant bedrock stratum at this site. What is the angle of bedding dip in the bedrock?
241.3 (SRP 2.5.4.2)	Provide a tabulation of the values of soil and rock design parameters for all significant foundation strata at this site. The table should include the range of design parameters considered reasonable for use in both static and dynamic analyses and indicate values actually adopted in design.
241.4 (SRP 2.5.4.2)	Provide plots of dynamic shear modulus and damping ratio as a function of shear strain for all significant foundation strata at this site. Indicate data points established by testing and the recommended design curve. Also, show the upper and lower bounds of these parameters used in your SSI analysis and discuss the basis for selection of these bounds. Show that the recommended design values are compatible with the estimated strains during an SSE event.
241.5 (SRP 2.5.4.2)	Table 2.5-20 is not self-explanatory. Add headings for the columns and notes as needed to clarify the table.
241.6 (SRP 2.5.4.2)	Provide dry density and moisture content data, and shear strength plots for the laboratory tests presented in FSAR Tables 2.5-25, 2.5-26 and 2.5-27.
241.7 (SRP 2.5.4.5)	Provide a plan showing the limits of excavation for foundations of the main plant structures including limits of cut slopes in both rock and overburden material and the bottom elevations of the excavations.
241.8 (SRP 2.5.4.5)	Provide details of blasting used in excavating the rock (e.g. type, patterns, etc) in order to understand if there were any effects on nearby structures and foundation rock integrity. What is the value of the peak particle velocity specified to control the blasting operation? How was it monitored and what are the results of the monitoring program? Provide details of the rock dowels used to stabilize vertical cut in bedrock. Was the rock broken-up due to blasting? Confirm that the bedrock left inplace where you have not used rock dowels was reasonably intact after excavation.

241. 9 (SRP 2.5.4.5) Section 2.5.4.5.2 of the FSAR states that the structural fill beneath Category I floor slabs does not support the slab but only acts as a construction support form. Provide detailed cross-sections and plans of such areas to show how the slab load is transferred to the bearing stratum. How is this slab and its support considered in the model for SSI analysis?

241.10 (SRP 2.5.4.5) For the sand-gravel fill used beneath structures and other soil backfill, which may contain from 0 to 10 percent fines, the specified compaction control requires the modified Proctor density criterion. Did the compaction specification permit use of or comparison with the relative density criterion? If so, provide the details of such specification and comparison.

241.11 (SRP 2.5.4.5) Provide a plan, longitudinal profile and cross-sections of the seismic Category I electrical duct line (907 and 922) and manhole (No. 1 CL-CE) which are indicated to be founded on compacted fill. Provide results (tabular or graphical form) of quality control tests performed to verify that the compacted fill is placed in accordance with the specifications.

241.12 (SRP 2.5.4.10) Provide the following information for each Seismic Category I structure in tabular form:

- Actual maximum bearing pressures imposed under both static and dynamic loading conditions.
- Description of the foundation materials (e.g., rock classification), and maximum allowable bearing pressures under static and dynamic loading conditions and the resulting bearing capacity factors of safety.
- Estimated displacement of the bedrock as a result of rock squeeze and rock swell. What is the estimated displacement along the fault plane, occurring beneath seismic Category I structures, as a result of an SSE event?
- Estimated total and differential settlements.
- Actual total and differential settlements and lateral movements monitored to date and drawings to indicate the locations where measured.

241.13 (SRP 2.5.4.10) Provide the following information needed for evaluating the lateral earth and water pressures on seismic Category I structures.

1. Cross-section of the backfill area showing:

   - the outer wall of the reactor building and inner face of the excavation in both rock and overburden

material. Show excavated slopes and the dimension of the space between the inner face of the excavation and the reactor wall.

- Show the limits of the backfill materials, both regular and special (vermiculite etc.), as placed above and below the top of bedrock. Is there a backfill material between the special backfill material (vermiculite etc.) and the face of the excavation in bedrock?

2. Plan showing the limits and location where you have used different backfill materials such as: vermiculite, vermiculite concrete, Rodoforam II, Nufoam type I, Vermiculite-Sand and regular backfill material.

3. Specifications for the gradation, placement and compaction control for various types of soil backfill material and special backfill materials. Present results of quality control tests performed to verify their compliance with the specifications.

4. FSAR Figure 2.5-106 shows that the vermiculite material responds like a very stiff material during the reload phase of compression. Have you investigated the response of other special backfill materials (presented in FSAR Figures 2.5-107 through 2.5-111) in their reload phase? Have you assumed in your analysis that these materials are in virgin compression phase and if so what precautions have you taken to assure that these are always in the virgin compression phase?

5. The coefficient of lateral earth pressure at-rest used in your analysis (K_o = 0.44) is less than what is normally used for compacted granular material. Provide the basis for the K_o parameter used in your analysis.

6. How has the magnitude of the inward movement of the excavated face of the bedrock, as a result of residual horizontal stresses and faults, been incorporated into your lateral earth pressure analysis?

7. What is the magnitude of the deformation of the backfill material during an SSE event? How is this considered in lateral earth pressure computations?

8. You have computed lateral earth pressures using the dynamic active earth pressure approach and then used it in the design as a lateral earth pressure on rigid walls under dynamic loading conditions. Justify your approach.

9. Provide a typical computation showing how you have established the vertical and horizontal limits of the sand-vermiculite backfill, which is used only at a few locations.

241.14 (SRP 2.5.4) The FSAR does not provide information on the Main Stack which is a seismic Category I structure. Provide detailed information on this in accordance with the R.G. 1.70 and SRP 2.5.4 to enable a safety evaluation by the staff. Also, provide data on the foundation materials supporting the electrical duct line which connects the Main Stack to the reactor building complex.

241.15 (SRP 2.5.4.6) Provide the following information on the permanent dewatering system.

1. Plan, cross section, typical details, and description of the permanent dewatering system installed around the reactor building.

2. Description of inspection and monitoring procedures to be required during years of plant operation.

3. The permanent dewatering system would appear to cause an increase in vertical uplifting pressure differences in the rock beneath the reactor building foundation. What effect does this have on the residual stresses in the bedrock and on the reactor building design? Discuss how this uplift pressure has been considered in your analysis?

4. In the event the dewatering system is not in operation, the ground water table would raise from elevation 164.0 ft to its normal elevation of 255.0 ft. What is the magnitude of the swell and rebound (due to stress relief) of the bedrock caused by this raise in the water table? Discuss in detail how this swell and rebound effect is considered in the foundation design of the structures and connecting conduits. and pipes.

241.16 (SRP 2.5.4.1 2.5.4.6 2.5.4.10) Provide updated records of piezometers, settlement monuments, inclinometers, extensometers, and linear displacement sensors in graphical form. Discuss your findings and conclusions on the significance of this data and any bearing on the past and future performance of the seismic Category I facilities at the site.

241.17 (SRP 2.5.5.2 and 2.5.5.4)

Provide the following information on the revetment-ditch system along the shore front.

1. Results of static and dynamic (SSE) stability analyses.

2. Results of quality control tests performed to verify that the as-built structure is in compliance with the specifications. Present the results in either a tabular or graphical format.

3. Provide details and discussion of the in-service monitoring of this safety-related structure including the scope and frequency of inspections.

241.18 (SRP 2.5.5)

The PMP-flood control berm at the project site is a safety-related structure but the FSAR does not present details except for a location plan. Provide detailed information on this structure in accordance with R.G. 1.70 and SRP 2.5.5 to enable a safety evaluation by the staff.

241.19 (SRP 2.5.4.5)

Have you used any other backfill material (such as lean concrete, mudmat, porous concrete, field concrete etc.) beneath and around seismic Category I structures? If so, provide a brief discussion on their specification, strength and placement criteria. Present results of quality control tests performed to verify their compliance with the specifications.

241.20 (SRP 2.5.4)

Provide drawings showing longitudinal profile and typical cross section of the intake tunnels and intake structures. Discuss your program to inspect these seismic Category I structures.

REQUEST FOR ADDITIONAL INFORMATION
NINE MILE POINT UNIT 2

MATERIALS ENGINEERING

251.1 Appendices G and H, 10 CFR Part 50 were revised in the Federal Register on May 27, 1983 and became effective on July 26, 1983.

a. Identify ferritic reactor coolant pressure boundary materials that do not comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50.

b. For materials, which cannot meet the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50, provide alternative fracture toughness data and analyses to demonstrate their equivalence to the requirements of 10 CFR Part 50.

c. To demonstrate conformance to Appendices G and H, 10 CFR Part 50;

(1) Provide pressure temperature limit curves for hydrostatic pressure and leak tests, heat-up, cooldown and core operations.

(2) Identify the withdrawal schedule, lead factor, test samples and materials in the Reactor Vessel Materials Surveillance Program.

(3) Indicate the reference temperature for materials in the reactor vessel closure flange region.

251.2 (5.2.3) Describe the degree of compliance to NUREG 0313, Rev. 1 for the balance of plant (i.e., exclusive of the NSSS).

251.3 (4.5.2) For welded austenitic stainless steel reactor internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions, demonstrate that components which may have carbon content at the maximum limit are not subject to intergranular stress corrosion cracking.

Auxiliary Systems
Request for Additional Information
Nine Mile Point Unit 2

410.4 (3.4.1) For those nonseismic Category I vessels, pipes and tanks located outside of buildings, discuss the effect of failure of these items during an earthquake and any potential flooding of safety-related structures, systems and components which may result. Provide a similar discussion of flooding potential from failure of non-tornado protected vessels, tanks and piping as a result of tornado wind loading or tornado missile strike.

410.5 (3.4.1) Provide a discussion of the protection afforded safety-related systems and components, including underground cables, with respect to flooding or wetting caused by (1) groundwater, (2) design basis flood, or (3) design basis precipitation.

410.6 (3.4.1) Provide a discussion of the design of plant features used to prevent flooding of redundant safety-related equipment inside buildings. A layout drawing should be provided to clearly identify the means of protection, separation of redundant equipment, and the height of any barrier which does not join the ceiling. Verify that all wall penetrations between redundant safety-related trains are watertight or justify your design.

410.7 (3.4.1) Verify that all safety-related building construction joints below grade have watertight seals, and all penetrations below the maximum anticipated groundwater level are watertight.

410.8 (3.5.1.1) (3.5.1.2) Provide a discussion of the results of an analysis which identifies the protection afforded redundant safety-related equipment from missiles generated by nonsafety-related sources. The analysis of the consequences of such an event should verify that a concurrent single active failure will not have an impact on plant shutdown. Consider potential missiles generated by nonsafety-related pressurized and rotating sources inside protective structures which house safety-related equipment. List and describe the size and energy of these nonsafety-related missile sources which can impact safety-related equipment and identify the method of protection for the redundant safety-related equipment.

410.9 (3.5.1.1) (3.5.1.2) Provide the results of an analysis and discuss protection provided against failure of high pressure gas bottles and accumulators which can result in potential internally generated missiles (both inside and outside of containment) that may impact on redundant safety-related equipment.

410.10 (3.5.1.1) (3.5.1.2) Provide the results of an analysis for each rotating component (pumps, turbines, and fans) which verifies that if an internal missile were generated, the casing would be capable of retaining the missile. For each rotating component whose casing cannot retain the internally generated missile and the missile could damage safety-related equipment, provide (1) a discussion of the methods used to protect the safety-related train and its redundant counterpart and other safety-related structures, systems and components in the path of the missile and (2) a drawing showing the component, missile paths, means of protection for other equipment, and the redundant safety-related train. This applies to both inside and outside containment.

1. Verify that no secondary missiles will be generated from any internally generated missile.

2. Verify that any internally generated missile from safety-related equipment will not affect the redundant safety-related train.

3. Provide the basis for your concluding that "...other rotating components..., such as fans, turbines, motors and compressors are not considered credible missiles."

410.11 (3.5.1.1) (3.5.1.2) Identify and provide a discussion of the effects of potential gravitational missiles and the methods used to protect safety-related structures, systems and components from these missiles (both inside and outside containment). Alternatively, verify that no gravitational missile is postulated because components over all redundant safety-related equipment are:

1. designed and installed to seismic Category I criteria (Regulatory Guide 1.29), or

2. supported to seismic Category I requirements (i.e., cannot fall as the result of the safe shutdown earthquake but may not be operational), or

3. located such that they cannot become a gravitational missile anywhere that may affect structures, systems or components which are safety-related.

410.12 (3.5.2) Provide a list of all safety-related components which are located outdoors or are exposed to tornado generated missiles. Describe the protection afforded each of these components to prevent their being damaged by tornado generated missiles in accordance with the guidelines of Regulatory Guide 1.117, "Tornado Design Classification." Include in this list a description of all HVAC system air intakes and exhausts including the protection afforded safety-related equipment near these openings, as well as emergency

diesel intakes and exhausts, doors, manholes, other openings in structures housing safety-related equipment, and any exposed piping. Identify the locations of these components on plant arrangement drawings.

410.13 (3.5.2) Provide drawings which show the physical routing of buried safety-related piping, the distance from the top of the pipe to grade, and discussion which verifies that the piping is sufficiently protected from tornado missiles.

410.14 (3.6.1) Provide the initial temperature, pressure and humidity conditions in the HPCI pump room, suppression chamber compartment and the steam vent compartment prior to the HPCI steam line break. Provide curves for each compartment of the temperature, pressure and humidity during the pipe break event. Provide the floor area for each compartment (square feet). Specify the calculated peak temperature, pressure and humidity for each compartment and the design environmental qualifications for safety-related equipment in each compartment.

410.15 (3.6.1)

a) Provide a list of all moderate energy systems with the normal temperature and pressure of each system.
Provide the analysis for moderate energy pipe breaks and their effects (flooding and jet impingement) in accordance with BTP ASB 3-1. Verify that a redundant train of safety-related equipment, component or cable will not fail as the result of a break in a dual-purpose moderate energy seismic Category I system.

b) Provide the results of a failure modes and effects analysis for high and nonseismic (nonsafety-related) moderate energy pipe failures which includes the single active failure in systems necessary to mitigate the consequences of the postulated pipe failure. For these cases, verify that safe shutdown can be achieved.

410.16 (4.6) Provide the information requested in our generic letter dated May 5, 1981, regarding the report entitled, "Safety Concerns Associated with a Pipe Break in the BWR Scram System," (NUREG-0803).

410.17 (4.6) Provide a point-by-point discussion of how each of the safety, design, operational, and surveillance criteria of the December 1, 1980 letter containing the "BWR Scram Discharge System Safety Evaluation" will be met by the Nine Mile Point 2 design.

410.18 (5.2.5) Provide a discussion of the reactor coolant pressure boundary (RCPB) leakage detection systems to include how each of the positions of Regulatory Guide 1.45 are met or justify any deviations. Provide information to show how total identified leakage flow rate is measured. Indicate the method used to obtain an accuracy of 1 gpm or better in one hour for unidentified leakage by the cyclic operation of sump pumps.

410.19 (5.2.5) Provide a discussion of intersystem leak detection methods including radioactivity, pressure, temperature, flow and pressure relief valve actuation indications for all systems connected to the reactor coolant system.

410.20 (9.1.1) Provide the specific K_{eff} values determined in your criticality analysis for the new fuel storage arrangement with the associated assumptions and input parameters. Clarify your assumption regarding water moderation when maximizing K_{eff}. Also, verify that the new fuel storage racks are capable of maintaining a K_{eff} of 0.98 or less under optimum moderation (foam, small droplets, spray or fogging) or identify the means provided for preventing such a condition in the new fuel storage vault.

410.21 (9.1.1) Provide a drawing of the new fuel storage facility.

410.22 (9.1.1) (9.1.2) Verify that nonsafety-related systems or structures which are not designed to seismic Category I criteria and are located in the vicinity of the new fuel storage facility and spent fuel storage facility will not fail in a manner that causes an increase in K_{eff} beyond the maximum allowable value.

410.23 (9.1.2) Verify that the spent fuel pool liner is seismic Category I, or that it is designed to remain in place and retain its leak tight integrity in a SSE. It is our position that the liner not fail in a manner which could result in mechanical damage to the spent fuel or result in an inability to maintain proper spent fuel pool cooling.

410.24 (9.1.2) Provide a discussion that compares the Nine Mile Point 2 spent fuel storage facility design against the criteria of ANS 57.2 paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2 and 5.3.4. Justify any deviations from the standards criteria.

410.25 (9.1.3) Verify that the spent fuel decay heat loads are based on NUREG-0800, Standard Review Plan, Section 9.1.3 and Branch Technical Position ASB 9-2.

410.26 (9.1.3) Verify that for the maximum normal heat load with normal cooling systems in operation, and assuming single active failure, the temperature of the pool will be maintained at or below 140°F.

410.27 (9.1.4) Provide a listing of all light loads (those of weight less than one fuel assembly) carried over the open reactor vessel or the spent fuel pool including their kinetic energy or impact with spent fuel and discuss the consequences of dropping of these loads on stored fuel. It is our position that dropping of these loads not result in release of radioactivity in excess of that assumed in the design basis fuel handling accident.

410.28 (9.1.5) Provide the information requested in the generic letter dated December 22, 1980, regarding conformance to the criteria contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Provide an analysis of the effects of dropping heavy loads other than the spent fuel cask or verify that load drops need not be postulated since the crane design meets the criteria of NUREG-0554, "Single Failure Proof Cranes." The analysis should satisfy the evaluation criteria of NUREG-0612, Section 5.1, and consider the consequences of dropping the reactor vessel head and vessel internals during preparation for or completion of fuel handling. In addition, the lower load block of both the containment building polar crane and the fuel building crane should be considered as a heavy load and an analysis of the consequences of their falling included in this evaluation. Alternatively, a comparison of heavy load handling equipment against the guidelines of NUREG-0554 should be provided. This comparison should

include a point-by-point evaluation against NUREG-0554 criteria, and a failure mode and effects analysis of the crane electrical system which confirms that no single failure will result in dropping of a load. An evaluation should also be provided which confirms that loss of one phase of a three phase power supply or phase reversal will not result in dropping of a load or other unacceptable load handling consequences. Also, verify that movement of spent fuel between units 1 and 2 will not follow a path that may endanger safety-related equipment such as essential buried piping or electrical cabling.

410.29 (9.2.1) Verify that failure of buried nonseismic Category I pipe in a SSE will not result in failure of safety-related buried pipes by soil erosion.

410.30 (9.2.1) Describe the design provisions in the service water system for component allowable operational degradation (e.g., pump leakage) and the procedures that will be followed to detect and correct these conditions when they become excessive.

410.31 (9.2.1) Your FSAR does not address the means provided to assure transfer of essential heat loads from the nonsafety-related Service Water System to the safety-related serive water system under accident conditions assuming the most limiting single failure. Provide this information.

410.32 (9.2.1) Provide a discussion of the design measures, procedures and operating practice employed to prevent fouling and degradation of the service water system as a result of marine life growth.

410.33 (9.2.6) Verify that the essential portions of the condensate storage facilities, including the isolation valves separating seismic Category I portions from the nonseismic portions are classified Quality Group C and seismic Category I.

410.34 (9.3.1) Verify that the instrument air system is designed in accordance with ANSI MC11.1-1976 (ISA 57.3). Discuss how the system complies with the criteria of this standard.

410.35 (9.3.1) Describe the means provided for assuring that instrument air quality is within the necessary limits to assure proper functioning of all air operated valves and instrumentation in safety-related systems.

410.36 (9.3.1) Discuss how compliance with the guidelines of Regulatory Guide 1.68.3, "Preoperational Testing of Instrument Air Systems," for each air system are met or justify any deviations.

410.37 (9.3.1) Provide a discussion of the maintenance and periodic testing program for each instrument air system to assure compliance with the requirements of ANSI MC11.1-1976. Specify the maximum time between testing of the compressed air system in the discussion.

410.38 (9.3.3) Provide an analysis to demonstrate that drainage of leakage water away from safety-related components or systems is adequate to assure safety functions for worst case flooding resulting from pipe breaks or cracks in high- or moderate-energy piping, or postulated failure in the most limiting nonseismic Category I piping near these safety-related components or systems when assuming a concurrent single active failure. The analysis must show that drainage by natural routes such as stairwells of equipment hatches or by nonseismic Category I drainage systems under failed conditions is adequate to prevent the loss of function of safety-related components and systems. Indicate how interconnected drains serving redundant safety-related equipment or cubicles are prevented from allowing leakage from one failed redundant train from backflowing and flooding out the other train. In those cases where separate drains are provided for redundant safety-related components or systems, provide an analysis that demonstrates that the compartment and/or area drains serving these components or systems have been sized for maximum leakage flow conditions.

It is our position that unless drainage capability by natural or by failed nonseismic Category I drainage systems can be demonstrated that you provide the following for all areas housing redundant safety-related equipment:

1. Leak detection sumps shall be equipped with redundant safety-related alarms which annunciate in the control room. Verify that if operator action is required on receipt of the alarm that flooding of redundant safety-related equipment will not occur within 30 minutes; <u>OR</u>

2. Provide separate watertight rooms and independent drainage paths with leak detection sumps for each redundant safety-related system.

410.39 (9.3.5) Discuss the protection provided the Standby Liquid Control System from missile and tornado effects.

410.40 (9.4.1) You have stated in FSAR Section 9.4.1.2 that two of the four control room ventilation system outdoor air intakes are tornado missile protected. Verify that the other two are also tornado missile protected, or explain why they need not be.

410.41 (9.4.1) Describe the capability for assuring a proper control room environment during long-term pressurization of the control room area following isolation from the outside intakes due to high radiation, chlorine or smoke, assuming the most limiting active failure. Indicate the maximum control room temperature under these conditions and justify the assumed limit. Discuss the means for assuring that the as-built control room ventilation system can provide the above assumed maximum design temperature.

410.42 (9.4.1) Describe the safety classification code and quality assurance criteria for the chlorine smoke and radiation detectors on the intake duct of the control room area ventilation system. It is our position that since these redundant detectors are vital for quick isolation of the control room ventilation system, these detectors be seismic Category I as recommended by Regulatory Guide 1.29, Position C.1.

410.43 (9.4.2) Your FSAR does not provide a discussion of the Spent Fuel Pool Area Ventilation System in accordance with NUREG-0800 Standard Review Plan, Section 9.4.2. Please provide such a discussion.

410.44 (9.4.3) Provide a description of the ventilation systems and/or room coolers provided to maintain the environment for safety-related equipment in the reactor building and radwaste building within allowable equipment qualification limits during the loss of all nonsafety-related ventilation equipment. Assume a loss of off-site power and concurrent single failure in these HVAC systems and verify that this condition will not affect plant safety. Discuss each system individually. Describe the source of air, its expected maximum temperature, routing and exhaust system. Make clear how nonsafety-related system failures affect the safety-related systems functions. Relate your description to specific drawings you have made or will make available.

410.45 (9.5.1) Describe the methodology used to verify that proper separation (fire protection) is provided for the safe shutdown capability in accordance with Section 5.b of BTP CMEB 9-1. Provide area arrangement drawings showing the safe shutdown system (including cable routing) in order that we may review the separation design.

410.46 (9.5.1) Address the means provided for assuring the function of the safe shutdown capability when considering fire induced failures in associated circuits as discussed in Enclosure 4A

410.47 (9.5.1) Describe in detail the design capability of the alternate shut-down capability for achieving hot and cold shutdown in accordance with Sections 5.b and 5.c of BTP CMEB 9-1 (Parts III.G and III.L of Appendix R). This discussion should include the equipment

which provides the capability to perform various safe shutdown functions, all required support equipment, and the instrumentation available for monitoring shutdown.

410.48 (10.3.1) Verify that instrumentation and controls for the main steam isolation valves are safety-related.

410.49 (10.3.1) Verify that the structure which contains the main steam piping up to the main steam (turbine) stop valves, is seismic Category I. Furthermore, verify that no nonseismic Category I piping or components are located above the main steam piping and associated valves which could fall and damage the main steam piping during a safe shutdown earthquake.

410.50 (10.3.1) Verify that the main steam isolation valves, shut-off valves in connecting piping, turbine stop valves, and bypass valves can close against maximum steam flow and differential pressure.

410.51 (10.4.7) Verify that the motor operated main feedwater isolation valves are supplied by redundant class 1E power supplies. Verify that the air-assisted check valve fails closed on loss of air pressure.

ASSOCIATED CIRCUIT GUIDANCE

I. INTRODUCTION

The following discusses the requirements for protecting redundant and/or alternative equipment needed for safe shutdown in the event of a fire. The requirements of Appendix R address hot shutdown equipment which must be free of fire damage. The following requirements also apply to cold shutdown equipment if the licensee elects to demonstrate that the equipment is to be free of fire damage. Appendix R does allow repairable damage to cold shutdown equipment.

Using the requirements of Sections III.G and III.L of Appendix R, the capability to achieve hot shutdown must exist given a fire in any area of the plant in conjunction with a loss of offsite power for 72 hours. Section III.G of Appendix R provides four methods for ensuring that the hot shutdown capability is protected from fires. The first three options as defined in Section III.G.2 provides methods for protection from fires of equipment needed for hot shutdown:

1. Redundant systems including cables, equipment, and associated circuits may be separated by a three-hour fire rated barrier; or,

2. Redundant systems including cables, equipment and associated circuits may be separated by a horizontal distance of more than 20 feet with no intervening combustibles. In addition, fire detection and an automatic fire suppression system are required; or,

3. Redundant systems including cables, equipment and associated circuits may be enclosed by a one-hour fire rated barrier. In addition, fire detectors and an automatic fire suppression system are required.

The last option as defined by Section III.G.3 provides an alternative shutdown capability to the redundant trains damaged by a fire.

4. Alternative shutdown equipment must be independent of the cables, equipment and associated circuits of the redundant systems damaged by the fire.

II. Associated Circuits of Concern

The following discussion provides A) a definition of associated circuits for Appendix R consideration, B) the guidelines for protecting the safe shutdown capability from the fire-induced failures of associated circuits and C) the information required by the staff to review associated circuits. It is important to note that our interest is only with those circuit (cables) whose fire-induced failure could affect shutdown. Guidelines for protecting the safe shutdown capability from the fire-induced failures of associated circuits are provided. These guidelines do not limit the alternatives available to the licensee for protecting the shutdown capability. All proposed methods for protection of the shutdown capability from fire-induced failures will be evaluated by the staff for acceptability.

A. Our concern is that circuits within the fire area will receive fire damage which can affect shutdown capability and thereby prevent post-fire safe shutdown. Associated Circuits* of Concern are defined as those cables (safety related, non-safety related, Class 1E, and non-Class 1E) that:

*The definition for associated circuits is not exactly the same as the definition presented in IEEE-384-1977.

1. Have a physical separation less than that required by Section III.G.2 of Appendix R, and;

2. Have one of the following:

   a. a common power source with the shutdown equipment (redundant or alternative) and the power source is not electrically protected from the circuit of concern by coordinated breakers, fuses, or similar devices (see diagram 2a), or

   b. a connection to circuits of equipment whose spurious operation would adversely affect the shutdown capability (e.g., RHR/RCS isolation valves, ADS valves, PORVs, steam generator atmospheric dump valves, instrumentation, steam bypass, etc.) (see diagram 2b), or

   c. a common enclosure (e.g., raceway, panel, junction) with the shutdown cables (redundant and alternative) and,

      (1) are not electrically protected by circuit breakers, fuses or similar devices, or

      (2) will allow propagation of the fire into the common enclosure, (see diagram 2c).

EXAMPLES OF ASSOCIATED CIRCUITS OF CONCERN

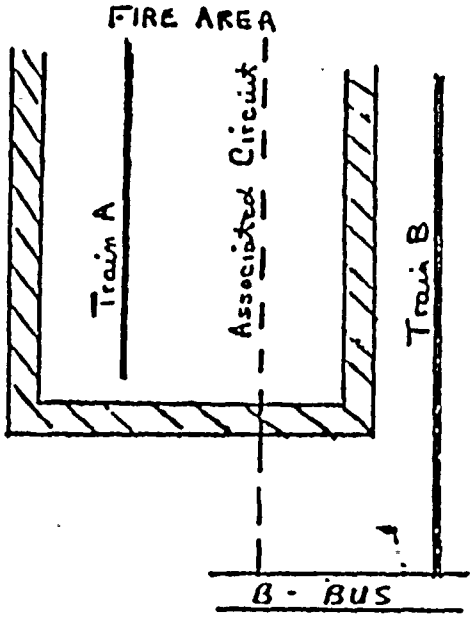
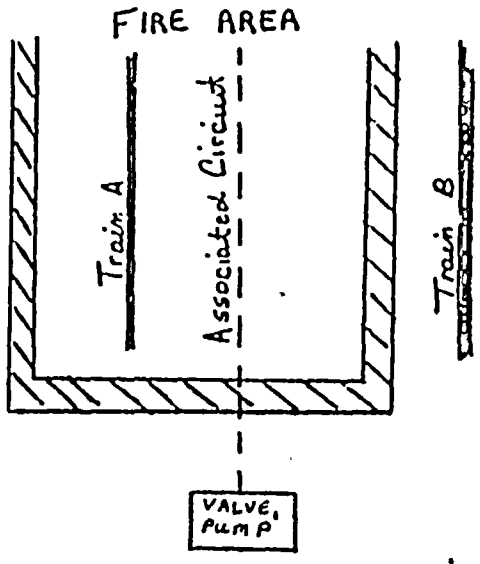
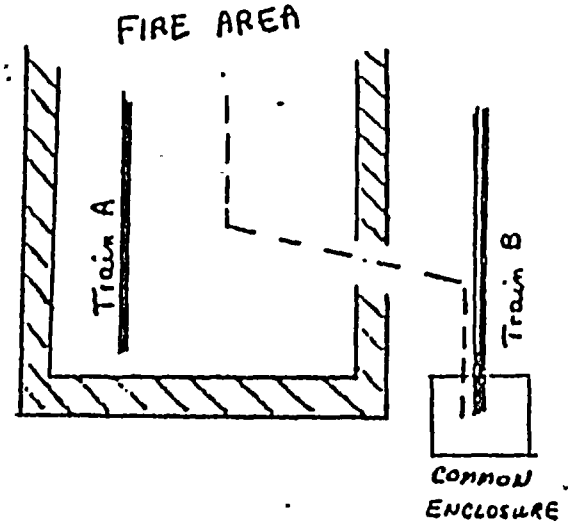


Diagram 2A



Equipment whose spurious operation could affect shutdown

Diagram 2B



The area barriers shown above meet the appropriate sub-paragraphs (a-f) of section III.G-2 of Appendix R.

Diagram 2C

B. The following guidelines are for protecting the shutdown capability from fire-induced failures of circuits (cables) in the fire area. The shutdown capability may be protected from the adverse effect of damage to associated circuits of concern by the following methods:

1. Provide protection between the associated circuits of concern and the shutdown circuits as per Section III.G.2 of Appendix R, or

2. a. For a common power source case of associated circuit:

   Provide load fuse/breaker (interrupting devices) to feeder fuse/breaker coordination to prevent loss of the redundant or alternative shutdown power source. To ensure that the following coordination criteria are met the following should apply:

   (1) The associated circuit of concern interrupting devices (breakers or fuses) time-overcurrent trip characteristic for all circuits faults should cause the interrupting device to interrupt the fault current prior to initiation of a trip of any upstream interrupting device which will cause a loss of the common power source,

   (2) The power source shall supply the necessary fault current for sufficient time to ensure the proper coordination without loss of function of the shutdown loads.

The acceptability of a particular interrupting device is considered demonstrated if the following criteria are met:

(i) The interrupting device design shall be factory tested to verify overcurrent protection as designed in accordance with the applicable UL, ANSI, or NEMA standards.

(ii) For low and medium voltage switchgear (480 V and above) circuit breaker/protective relay periodic testing shall demonstrate that the overall coordination scheme remains within the limits specified in the design criteria. This testing may be performed as a series of overlapping tests.

(iii) Molded case circuit breakers shall periodically be manually exercised and inspected to insure ease of operation. On a rotating refueling outage basis a sample of these breakers shall be tested to determine that breaker drift is within that allowed by the design criteria. Breakers should be tested in accordance with an accepted QC testing methodology such as MIL STD 10 5 D.

(iv) Fuses when used as interrupting devices do not require periodic testing. Administrative controls must insure that replacement fuses with ratings other than those selected for proper coordination are not accidentally used.

b. For circuits of equipment and/or components whose spurious operation would affect the capability to safely shutdown:

(1) provide a means to isolate the equipment and/or components from the fire area prior to the fire (i.e., remove power cables, open circuit breakers); or

(2) provide electrical isolation that prevents spurious operation. Potential isolation devices include breakers, fuses, amplifiers, control switches, current XFRS, fiber optic couplers, relays and transducers; or

(3) provide a means to detect spurious operations and then procedures to defeat the maloperation of equipment (i.e., closure of the block valve if PORV spuriously operates, opening of the breakers to remove spurious operation of safety injection);

c. For common enclosure cases of associated circuits:

(1) provide appropriate measures to prevent propagation of the fire; and

(2) provide electrical protection (i.e., breakers, fuses or similar devices)

C. INFORMATION REQUIRED

The following information is required to demonstrate that associated circuits will not prevent operation or cause maloperation of the shutdown method:

a. Describe the methodology used to assess the potential of associated citcuit adversely affecting the shutdown capability. The description of the methodology should include the methods used to identify the circuits which share a common power supply or a common enclosure with the shutdown system and the circuits whose spurious operation would affect shutdown. Additionally, the description should include the methods used to identify if these circuits are associated circuiys of concern due to their location in the fire area.

b. Show that fire-induced failures (hot shorts, open circuits or shorts to ground) of each of the associated circuits of concern will not prevent operation of cause maloperation of the shutdown method.

2. The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, the interface most likely consists of two redundant and independent motor operated valves. These two motor operated valves and their associated cables may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire initiated LOCA through the high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

a. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

b. For each set of redundant valves identified in a., verify the redundant cabling (power and control) have adequate physical separation as required by Section III.G.2 of Appendix R.

c. For each case where adequate separation is not provided, show that fire induced failures (hot short, open circuits or short to ground) of the cables will not cause maloperation and result in a LOCA.

NINE MILE POINT NUCLEAR STATION UNIT 2
REACTOR SYSTEMS
CHAPTER 15 - ACCIDENT ANALYSES

REQUEST FOR ADDITIONAL INFORMATION

440.41

(15.0) Regulatory Guide 1.70 states that a single failure and operator error requirement should be applied to all transient events discussed in FSAR subsection 15.0.2.

The Standard Review Plan recommends the use of conservative scram characteristics, e.g., a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

Demonstrate that the most limiting transient event in each category (FSAR 15.0.2) with the worst single failure or operator error does not violate MCPR and peak pressure limits. Consider conservative scram characteristics in your analyses.

Provide justification that the input parameters in Table 15.0.-3 are conservative.

440.42

(15.0) In the FSAR, generator load rejection without bypass and the turbine trip without bypass transients have been classified as infrequent events (FSAR Table 15.0.1).

The reclassification of these events has been under review by the staff and has not been approved. The staff requires that

these events be categorized as moderate frequency events. Therefore, the operating MCPR limit must be adjusted so that the safety limit of MCPR 1.06 is not violated by these transients.

440.43

(15.0) Structures, systems and components that provide reasonable assurance that a reactor facility can be operated without undue risk to the health and safety of the public are defined by the staff as "important to safety". This encompasses all equipment that contribute in an important way to safe plant operation and the protection of the public. This classification includes as a subset all structures, systems and components more specifically classified as "safety-related" or "safety-grade".

In your analysis of anticipated operational occurrences credit is assumed for equipment which has not been classified as "safety-related". However, this equipment is clearly important to safety since it is used to mitigate the consequences of plant transients. One example is the use of the main turbine bypass system to relieve pressure during the feedwater controller failure to maximum demand event. It is the staff position that all equipment which is important to safety which has not been classified as safety-related be

demonstrated reliable . Accordingly, the following information should be provided:

(1) A listing of all equipment which is not classified as safety-related but is assumed in FSAR analyses to mitigate the consequences of transients or accidents.

(2) Justification for the assumption of operability of this equipment based upon equipment quality, reliability, and proposed surveillance requirements.

(3) Discuss the consequences of those events concerning (i) number of fuel failures, (ii) Δ CPR and (iii) Δ peak pressure that would result if only safety grade systems or components were considered in the specific transients analyses taking credit of non-safety grade systems or components.

440.44

(15.0) Provide an analysis of the "Loss of Instrument Air" transient.

440.45

(15.1.1) The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the consequences of loss of feedwater heating in the manual flow control mode.

Provide a detailed description, including functions and time constant, for this scram system (STPT).

Describe the surveillance testing for this system. We require a provision of limiting operating condition and surveillance requirements in the Nine Mile Point Unit 2 Technical Specifications.

440.46

(15.1.1.) Operation of Nine Mile Point 2 with partial feedwater heating might occur during maintenance or as a result of a decision to operate with lower feedwater temperature near end of cycle.

We require analyses to justify that this mode of operation will not result in (1) greater maximum reactor vessel pressure than those obtained with the assumptions used in FSAR section 5.2.2, or (2) a more limiting MCPR than would be obtained with the assumptions used in FSAR chapter 15.0.

Otherwise, the staff will condition the license to prohibit operation in this mode.

Provide the basis for the maximum reduction in feedwater heating.

440.47

(15.1.2)

(15.3.3)

(15.3.4) Feedwater controller Failure - Maximum Demand, Recirculation Pump Seizure and Recirculation Pump Shaft Break operational transients analyses include the use of non-safety grade equipment, in particular the high water level (L8) trip and the turbine bypass system.

We require that either:

(1) The analyses be redone assuming no credit for these systems.

(2) That these systems be upgraded to safety-related status, or

(3) The high water level (L8) trip and the turbine bypass system be identified in the Technical Specifications with regard to availability, set points and surveillance testing.

44048 TMI Action Item II.K.3.44
Evaluation of Anticipated Transients with Single Failure to Verify no Fuel Damage.

Your discussion concerning this item is not sufficient.

Nine Mile Point Unit 2 must verify and confirm that the assumptions and initial conditions used in the BWR Owner's generic analyses are applicable or are bounding for this plant.

440.49

TMI Action Item II.B.1 Reactor Coolant Systems Vents.

Your response concerning this item is not completely acceptable.

Provide a discussion concerning RHR heat exchanger remote vent valves for post LOCA operation.

ACCIDENT EVALUATION

NINE MILE POINT UNIT 2, QUESTIONS

450.1 (6.5.3)

The SAR states that the SGTS will be able to produce and maintain subatmospheric pressure within the reactor building within one minute following its actuation. Criterion II.2 of SRP section 6.5.3 is that 0.25 inches of water gauge (70 pa) below atmospheric pressure must be maintained within those plant volumes that are to be considered as comprising a secondary containment. In order to assure that the SGTS functions as intended (i.e., produces a subatmospheric pressure of 0.25 inches water gauge), what surveillance and testing requirements are proposed for the secondary containment?

450.2 (6.7)(15.6.5)

Following any accident requiring containment isolation, a possible leakage path exists through the main steam line isolation valves. Such leakage could occur through both the valve seat and the valve stem packing. The FSAR contains no section 6.7 covering the subject matter, and sections 1.2.9.11 and Table 1.8-1,page 110 indicate because of the unique design of the positive seal of the MSIV ball valve, no seal system is required. Describe the plant design provisions that exist to prevent or collect main steam line isolation valve leakage.

450.3 (15.6.5)

Table 15.6-13 states that the reactor building recirculation rate is $7x10^5$ cfm. Section 9.4 of the FSAR lists the capacity of the reactor building ventilation system fans as 140,000 cfm (three 50% capacity 70,000 cfm trains). Following initiation of the SGTS, which fans are designed to run at what flow rates to achieve recirculation flow? Also, indicate the timing for automatic and/or manual initiation from the onset of the accident.

RADIOLOGICAL ASSESSMENT

Request for Additional Information
Nine Mile Point 2

471.1 Subsection 12.2.2.2.8 Effect of Liquid Radwaste Handling Areas,
12.2.I.2 last sentence (on page 12.2-15) of the FSAR states that: Expected airborne radioactivity levels in normally occupied areas are at or below ambient outside air concentrations. Provide justification for this statement.

471.2 Table 12.2-14, "Airborne Radionuclide Concentration in the Reactor
12.2.I.2 Building from a Main Steam Relief Valve Blowdown", states that this information will be provided in a later amendment. Provide this information.

471.3 Table 12.2-15,"Airborne Radionuclide Concentration in Liquid Radwaste
12.2.I.2 Handling Area due to Pump Leakage", states that this information will be provided in a later amendment. Provide this information.

471.4 Subsection 12.3.1, Facility Design Features, first sentence states
12.3.I.2.d that: "The design objectives and the design feature guidance given in Regulatory Guide 8.8 are incorporated into the Unit 2 plant to the extent discussed in Section 12.1.2."

Section 1.8, conformance to NRC Regulatory Guides, Table 1.8-2 Regulatory Guide 8.8, Revision 4 (March 1979 draft), states in Position: "See Section 12.1 and 12.5.3 for an assessment of this Regulatory Guide."

The acceptance and implementation of the Regulatory Guide 8.8 by NMP-2 is evident throughout the FSAR, especially in Chapters 11, 12 and 13. The position in Chapter 1.8, Table 1.8-2 should be revised to reflect this in a broad acceptance statement by stating, in effect, that NMP-2

considered all ALARA features recommended in the Regulatory Guide 8.8, Rev. 4 for the design and operation of the NMP-2, and the implementation of this guide is described in the FSAR, in the Chapters 11, 12, and 13.

471.5 Subsection 12.3-1.2, Radiation Zoning and Access Control, in table
12.3.I.1.b "Zone Designation and Expected Maximum Occupancy, Zone V" states that this zone is classified Unrestricted Area-Occupational Access (50-67 hr/week) and has a design Radiation Dose Rate <2.mrem/hr**. This does not comply with 10 CFR Part 20.105(b)(2) or (a). This area must therefore be classified as a restricted area, requiring compliance with 10 CFR Part 20.202(a)(1), Personnel monitoring. An asterisk should be added to Zone V designation description, signifying that personnel monitoring equipment is required to be worn.

471.6 Subsection 12.3.1.2, Radiation Zoning and Access Control, states that:
12.3.II.5 High Radiation Areas are posted. Access is controlled by doors or gates and is permitted by Radiation Work Permit (RWP). Areas above 1,000 mrem/hr are kept locked, and access is controlled by control room supervisory personnel. Also, in the same subsection on page 12.3-11 in second paragraph, fourth sentence, reference is made to Shift Supervisors maintaining positive access control over High Radiation Areas (>1,000 mrem/hr) at all times.

**The dose rate of 1.5 mRem/hr has been used as a design value to ensure that the 2 mRem/hr requirement has been met. Two mRem/hr is used as the operational limit.

10 CFR Part 20,203(c)(2)(i), Caution signs, labels and controls, states in (2) "Each entrance or access point to a high radiation area shall be: (i) Equipped with a control device which shall cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 milllirems in 1 hour upon entry into the area, or ....., (iii) Maintain locked except during periods when access to the area is required, with positive control over each individual entry." To comply with the above, the value of 1,000 m em/hr should be changed to 100 mrem/hr.

471.7 12.3-12.4-I.1.c Subsection 12.3.2.2.3, Plant Shielding Description, on page 12.3-15, Main Control Room. Shield thickness of the main control room walls (2'-0") is shown on Figure 12.3-63. In addition, specify the shielding thickness above the control room (ceiling or equivalent shielding).

471.8 12.3-12.4-I.4 Subsection 12.3.4.2.5, Sensitivities and Ranges, states that: Each monitoring system has a minimum detectable concentration such that 10 MPC-hour (maximum permissible concentration as defined in 10 CFR 20) of airborne particulate and iodine radioactivity can be detected in any one compartment which has the possibility of airborne contamination.

Subsection 12.5.3.3.4, Respiratory Protection, states that when airborne radioactivity is detected in a restricted area at 25 percent of the maximum permissible concentration (MPC) limits set in 10 CFR 20, Appendix B, the area is isolated and posted as an airborne radioactivity area, and access is controlled.

Explain how the 25% of MPC will be detected.

471.9 Subsection 12.3.4.3, Accident Consideration, states that the plant is designed so that the main control room is the only area
12.3.II.1 that need be occupied during the course of an accident.

NUREG-0737 in Section II.B.2(3), Dose Rate Criteria, states in part that: "(a)...The control room and onsite technical support center are areas where continuous occupancy will be required." In addition, reactor coolant and containment gas sample stations and the radiochemical analysis laboratory will require infrequent access. Subsection 12.3.4.3 should be revised to comply with NUREG-0737 requirements.

471.10 Figure 12.3.32, Shielding Arrangement and Facilities Piping Tunnel Plan, and Figure 12.3-65, Radiation Zones Piping Tunnel Plan,
12.3.I.2 state that these figures will be provided in a future amendment.

Provide this information.

470.11 Subsection 12.4.2.1, Man-Rem Evaluation, states that the Man-Rem evaluation is in progress, the details of which will be included as an amendment to the Unit 2 FSAR.

Provide this information.

470.12 Subsection 12.4.2.2, Inhalation Exposure, states that the calculation of inhalation exposure is in progress, the results of which will be included as an amendment to the Unit 2 FSAR.

Provide this information.

470.13 Subsection 12.4.3.1, N-16 Dose Contributions, states in the second paragraph in the Major Shielding section that the dose rate from the Unit 2 turbine building N-16 direct and air scatter contributions at the nearest point of the RAB is being calculated and will be provided in an amendment to the Unit 2 FSAR.

Provide this information.

470.14 Table 12.4-1, Estimated Doses at Locations Outside the Plant Structures, states that the information to be presented in this table will be provided as an amendment to the Unit 2 FSAR.

Provide this information.

471.15 Provide the missing information in Table 12.5.3, Personnel Monitoring Instrumentation.

471.16 NUREG-0737, Section II.B.2 states, "Documentation Required: (4) The projected doses to individuals for necessary occupancy times in vital areas and dose rate map (post accident) for potentially occupied areas."

Provide this information.

471.17 Confirm that personnel exposure (reference Section 1.10 Unit 2 Response to Regulatory Issues Resulting from TMI) of less than 3 and 18 3/4 rems to the whole body and extremities, respectively, during the collection of post accident samples, is based on pressurized reactor water radiation sources of 100% noble gases, 50% iodines, and 1% particulates, as required by NUREG-0737, Section II.B.2.

471.18 Standard Review Plan, Section 12.5, Operational Radiation Protection Program, in paragraph VI, References, lists those Regulatory Guides which will be used by the staff in performing their review of SRP 12.5. The following Regulatory Guides, referenced in that section, should be added to FSAR Section 1.8:

R.G. 8.6 "Standard Test Procedures for G-M Counters"
R.G. 8.13 "Instruction Concerning Prenatal Exposure"
R.G. 8.14 " Personnel Neutron Dosimeters"
R.G. 8.15 "Acceptable Programs for Respiratory Protection"
R.G. 8.20 "Applications of Bioassay for I-125 and I-131"
R.G. 8.28 "Audible Alarm Dosimeters"

471.19 Provide the following:

(a) Item II.B.2- Provide a description of post-accident access and shield design review.

(b) Provide a description of the buildup of activated corrosion products (crud) in various components and systems. (see also question 492.1)

(c) Describe in Section 12.3 of the FSAR post-accident vital area monitors.

(d) Section 12.5, II.B.4- Include audible alarm dosimeters in FSAR Table 12.5-3. Personnel Monitoring Instrumentation and quantity should also be specified.

QUESTIONS ON NINE MILE POINT-2

Thermal Hydraulics Section

Core Performance

492.1 SRP4.4-II.8 Section 4.4 of the Standard Review Plan states that the crud effects should be accounted for in the thermal-hydraulic design by including it in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should be capable of detecting a 3 percent pressure drop in the RC flow. The staff found that section 4.4 of the Nine Mile Point 2 FSAR has not disscussed crud effects and the process monitoring provision. Provide a submittal addressing the crud effects. The assumptions used for amount of crud in design calculations and the sensitivity of operating limit MCPR and core pressure drop to variations in the amount of crud present should be addressed. However, provisions to detect crud build-up in the core are of no concern during power operation and need not be described because of the power-flow map characteristics showing a decrease in power as RC flow decreases.

492.2 SRP 4.4-II.3 The staff is performing a generic study of the thermal hydraulic stability characteristics of BWRs under normal operation, anticipated transients and accident conditions. The results of this study will be applied to the staff review for acceptance of stability analyses. In the interim, the staff concludes that past operating experiences and inherent thermal-hydraulic characteristics of BWRs provide a basis for accepting the stability evaluation for normal operation and anticipated transients. However, in order to provide additional margin to stability limits, natural circulation operation will be prohibited by the Technical Specifications. Any action resulting from the staff generic study will be applied to Nine Mile Point 2

492.3 SRP 4.4-II.3 Because the thermal-hydraulic stability analysis is for the first cycle only, new analytical results must be reviewed and approved by the staff prior to second cycle operation.

492.4 SRP 4.4 II.5 No analysis has been presented for the operating limit MCPR or thermal-hydraulic stability characteristics for one loop operation. One loop operation will not be permitted until supporting analyses are provided and approved by the staff.

492.5 Because of BWR fuel design changes (i.e., smaller rod diameter, increased gap conductance), the calculated thermal-hydraulic
SRP 4.4- decay ratios show a decrease in stability margin. Verify that decay
II.5 ratio as a function of power shown in Figure 4.4-6 of FSAR can also be applied to P8X8R type of fuel used in the Nine Mile Point-2 Core.

492.6 The FSAR includes a very limited description of the loose parts monitoring system (LPMS). The staff has reviewed the FSAR regarding
SRP 4.4- the LPMS program and found additional information is needed .
II.7 Table 4.4.0 attached, "Summary Review of Nine Mile Point 2 LPMS," lists the staff's findings on the areas conforming to Regulatory Guide 1.133 and areas where additional information (these with symbol I or NI) is required. The LPMS should be installed to meet the operability requirements of Regulatory Guide 1.133. The LPMS must be operational and capable of recording vibration signals for signature analysis at the time of initial startup testing. Revise the evaluation of conformance to Regulatory Guide 1.133 to include a discussion of the following areas:

1. A description and evaluation of diagnostic procedures used to confirm the presence of a loose part.

2. A description of how the operators will be trained in the purpose and implementation of the system.

3. A description of system calibration, including signature analysis, evaluation of background noise and alarm setttings.

4. A description and evaluation of alert level establishment procedure with consideration of internal and external background noises.

Provide a commitment to provide the additional information identified in Table 4.4.0 prior to power operation.

492.7 In order for the staff to reach a conclusion concerning the instrumetation requirements for detection of inadequate core
SRP 4.4- cooling (ICC), submit a plant-specific evaluation addressing the
II.9 Nine Mile Point 2 position with respect to the BWR recommendations in SLI-8211, "Review of BWR Reactor Water Level Measurement Systems" dated July 1982, for upgrading the existing water level instruments. If no modifications to the existing water level systems will be made, address the adequacy and reliability of the existing water level systems for responding to excessive drywell temperature, reactor depressurization, rupture of water level reference leg, failure of water level transmitter and set point trip mechanism drag as identified in SLI-8211.

The evaluation should also address the applicability of the BWROG findings in report SLI-8218, "Inadequate Core Cooling Detection in Boiling Water Reactors" dated November 1982, regarding the need for additional instrumentation for Nine Mile Point Unit 2.

TABLE 4.4.0

SUMMARY OF REVIEW OF NINE MILE POINT 2

RG 1.133 SECTION	LPMS
C.1 System Characteristics	
a. Two sensors at each natural collection region.	C
b. Sensitivity of 0.5 ft-lb within 3ft of sensor.	NI
c. Physical separation of instrumentation channels.	NI
d. Automatic data acquisition (tape recorder).	I
e. Automatic comparison of signal to an alert level.	NI
f. Periodic system operational verification and calibration.	NI
g. Ability to function after seismic event.	NI
h. Quality of system components.	NI
i. Ease of repair to minimize radiation exposure.	NI
C.2. Establishing the Alert Level	
a. Logic to recognize LP in presence of noise.	NI
b. Override of noise caused by control rod movement, etc.	NI
c. Alert level a function of plant operating conditions.	NI
d. Compensation for different background noise on sensors.	NI
C.3. Using the Data Acquisition Modes	
a. Manual Mode	
(1) Pre-op tests to establish alert level.	NI
(2) Startup and power operation.	
a. Submit alert level within 90 days after startup.	NI
b. Perform channel check each 24 hours.	NI
c. Listen to audio output each 7 days.	NI
d. Perform functional test each 31 days.	NI
e. Verify background noise each 92 days.	NI

RG 1.133 SECTION	LPMS
(3) Verify channel calibration each 18 months.	NI
b. Automatic data recording when alert level is exceeded	I
C.4. Content of Safety Analysis Reports	
a. Sensor type, location, mounting, and criteria for these.	NI
b. Description of data acquisition, recording, and calibration.	I
c. Major sources of extraneous noise.	NI
d. Quality assurance of data.	NI
e. Description of alert level determination and alert logic	NI
f. Reference to Technical Specification.	NI
g. Description of diagnostic procedures used to confirm loose part.	NI
h. Channel check procedures.	NI
i. Maintenance procedures to minimize radiation exposure.	NI
j. Training program.	NI
k. Verification that LPMS will function after a seismic event.	NI
C.5. Technical Specification for Loose-Part Detection System.	NI
C.6. Notification of a Loose Part.	NI

KEY: C - Conformance with RE 1.133.

NC - Nonconformance with RG 1.133.

I - Insufficient information provided.

NI - No information provided.

NA - Not applicable at this time.