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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

Attachments 1 through 15 to 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:

Subject	SRP Sections	Attachment
Seismology & Geology	2.3.1 - 2.3.5	1
Hydrologic Engineering	2.4.1 - 2.4.14	2
Inservice Inspection	5.2.4, 6.6	3
Quality Assurance	17.0	4
Environmental Qualification	3.11	5
Seismic & Dynamic Load Qualification	3.10	6
Fire Protection	9.5.1	7
Chemical Technology	5.4.8, 6.1.2, 9.1.3, 9.3.2 10.4.6	8
Site Analysis	2.1.1 - 2.2.3	9
Reactor Systems	6.3	10
Meterology	2.3.1 - 2.3.5	11
Effluent Treatment	11.1 - 11.5	11
Containment Systems	6.2.1 - 6.2.6	12
Reactor Physics	4.3, 15.4.1 - 15.4.3, 15.4.7, 15.4.9	13
Initial Test Programs	14.2	14
Generic Issues	--	15

OFFICE							
SURNAME							
DATE							

As discussed in our letter of July 25, 1983, additional requests for information will be transmitted to you as we complete our reviews of the remaining sections. Also, additional information may be requested in the above areas, specifically as it pertains to:

(a) Quality Assurance - Safety-Related Structures, Systems and Components (Q-List) Controlled by the QA Program.

(b) Meterology - cooling towers and environmentally-related questions

(c) Geosciences - complex fault investigations at the site

(d) Containment Systems - Hydrodynamic and Mark II containment loads

In addition, concerns related to ultrasonic testing in Attachment 3 should be resolved prior to beginning the preservice inspection program.

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,

Original signed by

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

Enclosures:
As stated

cc w/enclosures:
See next page

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Nine Mile Point 2

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Nine Mile Point 2
Seismology Questions

230.3 The staff's position has been that the largest historical earthquake, in terms of intensity, in the Central Stable Region which has not been associated with tectonic structure is the March 9, 1937 Anna, Ohio (maximum Modified Mercalli intensity VII-VIII) event. Using this controlling earthquake and following the Standard Review Plan, Section 2.5.2, results in a safe shutdown earthquake (SSE) characterized by a Regulatory Guide 1.60 design response spectrum with a high frequency anchor of 0.19g. This exceeds the SSE proposed for Nine Mile Point Nuclear Station Unit 2 in section 2.5.2.6 of the Final Safety Analysis Report.

In recent operating license reviews the staff has accepted the use of site-specific response spectra developed from earthquake strong-motion records of appropriate magnitude, distance and site conditions to characterize the response spectrum of the SSE. The staff has observed that the 1937 Anna, Ohio earthquake (magnitude (m_b) 5.0-5.3) along with other central United States events have similar magnitudes. Therefore, for the Nine Mile Point Nuclear Station Unit 2 site, a site-specific spectrum would be developed from a suite of strong motion records from magnitude 5.3 ± 0.5 earthquakes recorded at distances less than about 25 kilometers from the source at rock sites. It has been the staff's position that the appropriate representation of the response spectra as derived directly from real time histories is the 84th percentile level. A site-specific spectrum may be computed directly or spectra from other appropriate sites may be utilized (see for example those used for Perry, Wolf Creek and Limerick).

Considering the staff's position, demonstrate the adequacy of the SSE by comparing it to either the Regulatory Guide 1.60 spectrum anchored at 0.19g or a suitable site-specific spectrum.

Q230.4 Regulatory Guide 1.60 states that the vertical design response spectrum values should be 2/3 those of the horizontal design response spectrum for frequencies less than 0.25 Hertz, for frequencies higher than 3.5 hertz the two spectra should be the same, and the ratio should vary between 2/3 and 1 for frequencies between 0.25 and 3.5 hertz. The proposed vertical design response spectrum for Nine Mile Point Unit 2 (FSAR Figure 2.5-84) is 2/3 of the proposed horizontal design response spectrum at all frequencies. Provide a justification for the currently proposed vertical response spectrum.

230.5 In section 2.5.2.4 of the FSAR you assign a maximum Modified Mercalli (MM) intensity of VII to the Attica, N.Y. earthquake of August 12, 1929 and state this suggests a MM intensity of IV at the site from this type event if it were to occur at the closest approach (100 km) of the associated structure (Clarendon-Linden fault) to the site. Numerous seismological references report this earthquake as having had a maximum MM intensity of VIII or a maximum Rossi-Forel intensity of IX. Justify your downgrading of the intensity of this earthquake. Using the higher maximum intensity estimate the vibratory ground motion from this type event occurring at 100 km from the site. Determine if this will affect the seismic design for Nine Mile Point Unit 2.

230.6 Section 2.5.4.5.2 of the FSAR states that all major Category I structures are founded on bedrock except for a Category I electrical ductline and manhole which are founded on structural fill and that fill was also placed beneath certain Category I floor slabs. For all Category I structures founded on fill and/or soil determine if amplification of the vibratory ground motion estimated for the bedrock can occur and determine how large and at what frequencies this amplification will be.

Geology Questions

231.1 Figures 2.5-5 and 2.5-6 are cross-sections with stratigraphic columns that do not conform in groupings with the same units in Figure 2.5-7. Regional and Site Stratigraphic Columns. For example in Fig. 2.5-7, the Oswego Sandstone is a formation in the Lorraine Group while in Fig. 2.5-56, the Oswego lies above the Lorraine. Please correct or explain the inconsistency.

Is the Tribes Hill Formation part of the Beekmantown (Fig. 2.5-7)?

231.2 On Fig. 2.5-7, please provide the rock type or types for all units named as "Formation", and for all Group names that do not have formation subdivisions listed. What, for example, is the Theresa Formation, which is on both generalized regional cross-sections, Fig. 2.5-5, 6 and the Stratigraphic Column (Fig. 2.5-7) but is not mentioned in the text nor described anywhere?

231.3 Figure 361.26-1 of your response to question 361.26 and Figure 2.5-28 of the FSAR are essentially the same with more information on the FSAR diagram. Both show the locations of the high angle faults and related information. Please explain why Trench No. 2 on the earlier Fig. 361.26-1 is located at least 200 ft more northeasterly than on the later Fig. 2.5-28.

231.4 The lengths of the three high-angle faults, the Barge Slip, Drainage Ditch and Cooling Tower faults, as discussed on p. 2.5-54 of the FSAR, have not been conclusively determined.

a. Trench No. 2 beyond the northeast projection of the Cooling Tower fault appears too short to intersect the fault if there is the slightest deviation from a linear trend. Why was this trench not extended north and south, especially considering that the fault appears to change trend from a more northerly trend between Trenches 4 and 5 to slightly more westerly from Trench 4 to Trench 3 and Pit 1 according to Figure 2.5-28. If the fault turned slightly more west of Pit 1, the fault could be too far south to intersect the northwestern-most Trench along the Cooling Tower fault.

b. On two separate occasions, in answer to question 361.26 and in section 2.5.1.2.3 of the FSAR, the statement is made that it is difficult to determine precisely the east-southeast termination of the Cooling Tower fault. Since you postulate that this fault is the

analog of the Drainage Ditch fault and therefore probably of the same length, please explain why a trench could not be dug near or beyond the postulated southeastern terminus, to verify the assumption?

c. Please explain the logic, including geometrical, mechanical or physical principles, that justify the assumption stated in the last full paragraph on p. 2.5-54 that "the length of the Cooling Tower fault is not significantly different from the length of its analog," and that all the faults extend to the same depth.

d. Inasmuch as the determined length of the Cooling Tower fault is based on assumption and not proven, what evidence precludes the possibility that the normal displacement increases with depth, as would be the case if there were recurrent movement through time?

231.5 What is the resolution of the magnetometer, i.e. how large does a basement offset have to be in order to be detected?

231.6 Please explain the basis for the assertion (p. 2.5-56) that the similarity of homogenization temperatures between the vein minerals of the fractures and the quartz clasts of the rock indicates that the deformation occurred after diagenesis. The logic is not obvious inasmuch as quartz clasts originate as igneous, or metamorphic minerals long before sedimentation, diagenesis, brittle fracture and vein mineralization.

231.7 On p. 2.5-60 the absence of mineralization in association with the buckling and reverse-slip displacement on the high-angle faults is used to deduce the near-surface origin of the structures and, by inference, their more recent development. Could the absence of mineralization be the result of continuous creep movements up to the present in a compressional regime which does not permit voids or extension along the structures?

231.8 On p. 2.5-62, discussion of the mechanism of reverse-slip displacement is attributed to the secondary effect of buckling instability which is associated with bedding-plane slip. Explain why all the deformation discussed here may not be attributed to bedding-plane slip in the form of creep that may be an on-going process.

231.9 What is the greatest depth at which bedding-plane slip has been detected at or near the site?

231.10 Mechanical theory of buckling (p. 2.5-63)

a. Summarize the reasons why the initial "deflection" in your discussion of the mechanical theory of the origin of the buckles is not simply drag effects of the earlier normal fault displacement on the high-angle fault and therefore not related to the buckling superimposed on the earlier structures.

b. Considering that greater stress is required to cause a new fracture in rock than to cause slip along a pre-existing fracture, explain how, in going from the amplification to rotation stages of buckling, a new fracture is formed at almost the same angle as the pre-existing one. Also discuss the mechanical considerations that would allow a shear fracture to develop at such a high angle (considering the Coulomb-Mohr-Navier theory of shear fracture).

231.11 The absence of evidence for reverse-slip deformation and buckling on the Barge Slip fault is attributed to its southerly dip, which is the only difference mentioned between this fault and the two parallel structures, the Cooling Tower and Drainage Ditch faults. The explanation is conjectural and not convincing. There is, however, another difference, which is the presence between the latter two faults of a low-angle thrust with relatively young movement. Evaluate the likelihood that the reverse-slip and concomitant buckling of the Cooling Tower and Drainage Ditch faults are the result of tangential stresses exerted on the faults by movements of the Radwaste Thrust structure.

231.12 Among the conclusions reached concerning the Cooling Tower fault, it is stated that minor subsurface adjustments may occur within the zone of buckling at a depth within the transition zone (50-200 ft). It is also asserted that these adjustments will not reach the surface because voids in the rock must first be closed. What estimates have been made as to how much movement (vertical or horizontal) may occur? What calculations

have been made to determine the minimum vertical movement necessary to close the voids and then reach the surface? Also provide the justification for the estimates.

231.13 In discussing the seismogenic potential of the Cooling Tower fault (p. 25-68) the assertion is made that the buckle fold is now locked and therefore unable to generate vibratory ground motion. However, this does not preclude the possibility of reverse fault motion on either of the faults bounding the rotated sliver. As your interpretation about downwarping as the initiator of the fault movement is conjectural, discuss the possibility of strain buildup on the Cooling Tower fault by creep on bedding planes and the possible resultant seismic effects of movement on either or both faults.

231.14 p. 2.5-72 - On the first line of the 2nd paragraph in the section on Thrust Faults reference is made to Fig. 2.5-29 in describing where the faults have developed. However, Fig. 2.5-29 is a stratigraphic column of the overburden and not relevant to the subject matter as discussed. Please send the figure you are referencing if it is missing, or correct the reference in the text.

231.15 It is stated that the walls of the intake and discharge tunnels, which have exposed Radwaste-type low angle thrust structures, will be bare and filled with water, with free-standing pipes within. What will be the effects of the long-term exposure of the faults to water on their stability. Is it likely to cause slip along planes of weakness (bedding or faults), locally?

231.16 It is not clear from your discussions about the various structures at the site, what is concluded about the age of the bedding-plane slip which is so prevalent. Although the discussions of both the Cooling Tower high angle fault and Radwaste Thrust fault attribute a role in their formation or deformation to bedding plane slip, the age of the bedding plane slip is not clearly stated, although the implications are that the slip is Pleistocene, either pre-Wisconsinan or related to the Lake Iriquois stage. Please expand your discussion of the bedding-plane slip at the site and in the site region. Include in your discussion but do not limit it to:

(a) the significance of Fig. 3-3 of Appendix 2I which shows bedding plane slip surfaces with a northwestward (updip) sense of movement, being truncated by the Paleozoic high angle faults of the Demster Fault zone;

(b) possible causes of the bedding plane slip.

(c) the possibility that the slip is Paleozoic in origin but renewed movement on preexisting weak gouge and brecciated surfaces provided the paths for glacial meltwater.

(d) In what ways the bedding plane slip may have influenced the structures to which they are related.

231.17 (a) Please provide a plot of the temperature-depth relationships of the minerals used to determine age of the joints and other structures. As there may be differing opinions concerning stability fields, etc., include a discussion justifying your choice of plot in depth determinations.

(b) The conditions of formation of the minerals within the faults or joints, not the joint itself, is postulated on the basis of homogenization temperatures. It is clear from some of the photos in Vol. I of the Fault Investigation (1978) that some of the structures formed in a compressional environment (subhorizontal slickensides) and experienced extension later (mineral coating with 3-dimensional undeformed crystals on the fracture surface). Please comment on the suggestion that the vein minerals have recorded changing stress regimes and suggest continuous or renewed movement through time on preexisting structures, rather than the conditions of formation of the faults, joints, slip planes, or voids.

HYDROLOGIC ENGINEERING QUESTIONS

NINE MILE POINT UNIT 2

240.10 (FSAR 2.4.2.3) (SRP 2.4.2)

The FSAR states that the roof drainage system is designed for the PMP rate of 8.4 inches per hour.

a) Provide a discussion of how this conclusion was reached. Include in this discussion your assumptions regarding credit taken for roof drains, the amount of plugging assumed and etc. State the height of parapets (if any) which may tend to pond water if the drainage system should become blocked by debris. Also, describe the dimensions and placement of scuppers or other openings (if any) that will tend to limit the depth to which water can pond.

b) Local PMP rainfall rates as determined from NOAA Hydrometeorological Reports Nos. 51 and 52 are considerably higher than 8.4 inches per hour for short durations. Discuss the effect of this higher rainfall rate on the roofs of safety related structures. Also, discuss the structural ability of the roofs to handle the resulting loads.

240.11 (FSAR 2.4.2.3) (SRP 2.4.2)

a) The FSAR states that the time of concentration for the local drainage basin at the plant site is one hour. Described how the time of concentration was determined.

b) Discuss in more detail the determination of maximum water levels for the PMP on the plant site drainage system. State which culverts or bridge sections were assumed to be plugged and which were not, and give the dimensions and elevations of all culverts or openings which had an effect on water level. Also, provide justification for your assumption regarding partially blocked or open culverts.

c) Estimate the maximum water level of the flood resulting from the PMP as determined from NOAA Hydrometeorological Reports, 51 and 52. If the resulting water level is above the entrances to Safety Related Buildings describe procedures to prevent ingress of water into those buildings.

240.12 (FSAR 2.4.7) (SRP 2.4.7)

Provide an analysis of maximum ice force on the safety related intake structures. Determine ice forces from both wind blown ice and from thermal expansion of solid ice sheets. It is recommended that the mechanical properties of the ice be used to determine limiting values for the ice forces rather than a statistical analysis of meteorological parameters responsible for causing the forces.

240.13 (FSAR 2.4.7) (SRP 2.4.7)

The FSAR states that frazil ice formation on the intake bar racks will be precluded by electrical heating elements. State the lowest temperature for which the heating elements are designed to keep the racks ice free and describe the data base and procedures used to estimated this temperature.

240.14 (FSAR 2.4.10) (SRP 2.4.10)

The FSAR states that the exterior flood protection berms and the revetment ditch system protect the plant from meteorologically induced probable maximum floods. The FSAR, however, does not discuss the seismic design criteria for either structure or provide an analysis of the effects of an earthquake combined with a lesser flood then the PMP. Provide the seismic design criteria for each structure and provide a discussion of the consequence of the following combinations of events:

a) 25 year flood on Lake Ontario and an SSE

b) Standard Project flood or equivalent on Lake Ontario and an OBE

c) 25 year rainstorm on site and an SSE

d) One half PMP on site and an OBE

240.15 (FSAR 2.4.13) (SRP 2.4.12)

It is not clear from either FSAR section 2.4.13.5 or from Fig. 2.5-110 as to what water level was used for determination of maximum subsurface hydrostatic loading on safety related structures and what water level was used for determination of combined loads including other independent environmental events such as earthquakes. Please clarify these points in response to this question.

240.16 (FSAR 2.4.2.3) (SRP 2.4.2)

Provide a map showing the elevations along the relocated Lake Road adjacent to Nine Mile Point Units 1 and 2 as well as the topography of the area to the south of the road. The area and detail shown should be sufficient to verify the water levels and flows over the road shown in Figure 2.4-1.

240.17 (FSAR 2.4.5) (SRP 2.4.5)

It is our understanding that the main stack in the northeast corner of the site is designed to withstand wind induced water waves which will be uneffected by the revetment-ditch system. Provide details of the analysis including wave heights, periods, bathymetry and the resulting wave force distribution against the stack.

NINE MILE POINT UNIT 2

Request for Additional Information
Review of the Preservice (PSI)/
Inservice (ISI) Inspection Program

250.1 To provide our input to SER Sections 5.2.4 and 6.6, the staff requires that the PSI Program Plan be submitted for review. The FSAR indicates that PSI examination of the reactor vessel will begin in 1985. Provide a schedule defining when the entire PSI Program will be completed and submitted for review. The PSI Program should include reference to the ASME Code Section XI Edition and Addenda that will be used for the selection of components for examinations, lists of the components subject to examination, a description of the components exempt from examination by the applicable code, and the examination isometric drawings.

Paragraph 50.55a(b)(2)(iv) requires that ASME Code Class 2 piping welds in the Residual Heat Removal Systems, Emergency Core Cooling Systems and Containment Heat Removal Systems shall be examined. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in IWC-1220. To satisfy the inspection requirements of General Design Criteria 36, 39, 42, and 45, the preservice inspection program must include volumetric examination of a representative sample of welds in the RHR, ECCS and Containment Heat Removal Systems.

Plans for preservice examination of the reactor pressure vessel welds should address the degree of compliance with Regulatory Guide 1.150. Discuss how the procedures to be used will be qualified to assure finding service-induced flaws on the ID surface.

250.2 Describe the measures taken to ensure that austenitic stainless steel piping welds, which have been determined to be "service sensitive" to IGSCC as defined in NUREG-0313, are examined using effective techniques and the methods of assuring adequate examination sensitivity over the required examination volume. Discuss the preservice examination criteria used to record, report and plot geometric or metallurgical ultrasonic indications in "service sensitive" piping systems to assure correlation of baseline data with inservice inspection results.

The ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978 and 1980 Edition specified the use of Appendix III of Section XI for ferritic piping welds. If this requirement is not applicable (for example, for austenitic piping welds), ultrasonic examination is required by Section XI to be conducted in accordance with the applicable requirements of Article 5 of Section V, as amended by IWA-2232. A technical justification is required if any alternatives are used. If Section XI, Appendix III, Supplement 7, will be used for the examination of austenitic piping welds, discuss the following:

1. All modifications permitted by Supplement 7.

2. Methods of qualifying the procedure for examination through the weld (if complete examination is to be considered for examination conducted with only one side access).

When using either Article 5 of Section V or Appendix III of Section XI for examination of either ferritic or austenitic piping welds, the following should be incorporated:

1. Any crack-like indication, regardless of ultrasonic amplitude, discovered during examination of piping welds or adjacent base metal materials should be recorded and investigated by a Level II or Level III examiner to the extent necessary to determined the shape, identity, and location of the reflector.

2. The Owner should evaluate and take corrective action for the disposition of any indication investigated and found to be other than geometrical or metallurgical in nature.

250.3 All preservice examination requirements defined in Section XI of the ASME Code that have been determined to be impractical must be identified and a supporting technical justification must be provided. The relief requests should include at least the following information:

1. For ASME Code Class 1 and 2 components, provide a table similar to IWB-2600 and IWC-2600 confirming that either the entire Section XI preservice examination was performed on the component or relief is requested with a technical justification supporting your conclusion.

2. Where relief is requested for pressure retaining welds in the reactor vessel, identify the specific welds that did not receive a 100% preservice ultrasonic examination and estimate the extent of the examination that was performed.

3. Where relief is requested for piping system welds (Examination Category B-J, C-F, and C-G), provide a list of the specific welds that did not received a complete Section XI preservice examination including drawing or isometric identification number, system, welds number, and physical configuration (e.g., pipe-to-nozzle weld, etc.). Estimate the extent of the preservice examination that was performed. When the volumetric examination was performed from one side of the weld, discuss whether the entire weld volume and the heat affected zone (HAZ) and base metal on the far side of the weld were examined. State the primary reason that a specific examination is impractical (e.g., support of component restricts access, fitting prevents adequate ultrasonic coupling on one side, component-to-component welds prevents ultrasonic examination, etc.). Indicate any alternative or supplemental examinations performed and method(s) of fabrication examination.

250.4 FSAR Section 6.6.8 states that no augmented inservice inspection will be required for Safety Class 2 and 3 systems and components since there is no Safety Class 2 or 3 high energy piping between containment isolation valves. Augmented inservice inspection should be performed on the welds of high energy piping in the containment penetration region where pipe breaks are not postulated and where the effects of the breaks can not be accommodated as described in FSAR Section 3.6.

REQUEST FOR ADDITIONAL INFORMATION

Nine Mile Point Unit 2

260.0 Quality Assurance

260.1 Describe the criteria for determining the size of the QA organization including the inspection staff, and present the projected number of professional QA/QC personnel to be on board during the operation phase recognizing this number will vary somewhat throughout the operations of Nine Mile Point Unit 2. Identify in more detail the QA and QC organizational positions on Figure 17.1-1. (1A5)*

260.2 Describe those provisions which assure that verification of conformance to established requirements is accomplished by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified. If this function is performed by individuals other than the QA organization, then identify the organizational position and the QA/QC qualification of that position. (1B2)

260.3 Describe those provisions which assure that designated QA and QC personnel, sufficiently free from direct pressures for cost/schedule, have the responsibility delineated in writing to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material. (1B4a)

260.4 Describe those provisions which assure that designated QA/QC individuals are involved in day-to-day plant activities important to safety (i.e., the QA/QC organizations routinely attend and participate in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignment throughout the plant and that there is adequate QA/QC coverage relative to procedural and inspection controls, acceptance criteria, and QA/QC staffing and qualification of personnel to carry out QA assignments). (1B6)

260.5 Describe the qualification requirements of the QA Manager which includes the following prerequisites: (1C2)

a. Management experience through assignments to responsible positions.

b. Knowledge of QA regulations, policies, practices, and standards.

c. Experience working in QA or related activity in reactor design, construction, or operation or in a similar high technological industry.

*This designation represents the particular items of Standard Review Plan Section 17.1 that the requests originate from.

The qualifications of the QA Manager should be at least equivalent to those described in Section 4.4.5 of ANSI/ASN-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," as endorsed by the regulatory positions in Regulatory Guide 1.8.

260.6 Identify the position responsible for the onsite QA/QC program and describe those provisions which assure this position has appropriate organizational responsibilities and authority to exercise proper control over the QA program. (1C3)

260.7 Describe those provisions which assure that the development, control and use of computer code programs associated with items important to safety will be conducted in accordance with the QA program and a description of how the QA program will be applied. (2A1c)

260.8 Describe or reference the QA program that will apply to the fire protection program.

260.9 Describe those QA program provisions which assure that NM will comply with 10 CFR Part 50, §50.55a.

260.10 Identify those existing or proposed QA procedures that require that Regulatory Guides listed in Section 1.8 of the SAR, and 10 CFR Part 50, §50.55a, will be met by document procedures. (2B4)

260.11 Provide a description that emphasizes how the docketed QA program description, particularly the Regulatory Guides listed in Table 17.0-1 of the SAR, will be properly carried out.

260.12 Provide a description of how management (above or outside the QA organization) regularly assesses the scope, status, adequacy, and compliance of the QA program to 10 CFR Part 50, Appendix B. These measures should include: (2C1)

a. Frequent contact with program status through reports, meetings, and/or audits.

b. Performance of an annual assessment preplanned and documented. Corrective action is identified and tracked.

260.13 Describe those provisions which assure that the indoctrination and training program includes the following: (2D)

a. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

b. Certificate of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria used to qualify personnel in each function.

260.14 Describe those provisions which assure procedures are established requiring a documented check to verify the dimensional accuracy and completeness of design drawing and specifications. (3E1).

260.15 In addition to the design controls specified in Section 17.1.3, describe those provisions which assure procedures are established requiring that design drawings and specifications be reviewed by the QA organization to assure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain the necessary QA requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results. (3E2)

260.16 Describe those provisions which assure guidelines or criteria are established for determining the method of design verification (design review, alternate calculations, or test). (3E3)

260.17 Describe those provisions which assure procedures are established for design verification activities which assure the following: (3E4)

a. The verifier is qualified and is not directly responsible for the design (i.e., neither the performer or his immediate supervisor). In exceptional circumstances, the designer's immediate supervisor can perform the verification provided:

(1) The supervisor is the only technically qualified individual.

(2) The need is individually documented and approved in advance by the supervisor's management.

(3) QA audits cover frequency and effectiveness of use of supervisors as design verifiers to guard against abuse.

b. Design verification, if other than by qualification testing of a prototype or lead production unit, is completed prior to release for procurement, manufacturing, construction or to another organization for use in other design activities. In those cases where this timing cannot be met, the design verification may be deferred,

providing that the justification for this action is documented and the unverified portion of the design output document and all design output documents, based on the unverified data, are appropriately identified and controlled. Construction site activities associated with a design or design change should not proceed without verification past the point where the installation would become irreversible (i.e., require extensive demolition and rework). In all cases, the design verification should be complete prior to fuel load for a plant under construction, or in the case of an operating plant, prior to relying upon the component, system, or structure to perform its fucntion.

c. Procedural control is established for design documents that reflect the commitments of the SAR; this control differentiates between documents that receive formal design verification by interdisciplinary or multi-organizational teams and those which can be reviewed by a single individual (a signature and date is acceptable documentation for personnel certification). Design documents subject to procedural control include, but are not limited to, specifications, calculations, computer programs, system descriptions, SAR when used as a design document, and drawings including flow diagrams, electrical single line diagrams, structural systems for major facilities, site arrangements, and equipment locations. Specialized reviews should be used when uniqueness or special design considerations warrant.

d. The responsibilities of the verifier, the areas and features to be verified, the pertinent considerations to be verified, and the extent of documentation are identified in procedures.

260.18 Describe those provisions which assure that the following are included if the verification method is only by test: (3E3)

a. Procedures provide criteria that specify when verification should be by test.

b. Prototype, component or feature testing is performed as early as possible prior to installation of plant equipment, or prior to the point when the installation would become irreversible.

c. Verification by test is performed under conditions that simulate the most adverse design conditions as determined by analysis.

260.19 Describe those provisions which assure that procedures are established to assure that verified computer codes are certified for use and that their use is specified. (3E4)

260.20 Describe those provisions which assure that responsible plant personnel are made aware of design changes/modifications which may affect the performance of their duties. (SRP Section 17.2.3, item 2)

260.21 Describe those provisions which assure that maintenance, modification and inspection procedures are reviewed by qualified personnel knowledgeable in QA disciplines (normally the QA organization) to determine: (SRP Section 17.2.6, item 2)

a. The need for inspection, identification of inspection personnel, and documentation of inspection results.

b. That the necessary inspection requirements, methods, and acceptance criteria have been identified.

260.22 Describe those provisions which assure that procedures are established for the review, approval, and issuance of documents and changes thereto and that the documents are reviewed for technical adequacy and inclusion of appropriate quality requirements prior to implementation. The QA organization, or an individual other than the person who generated the document but qualified in QA, reviews and concurs with quality affecting documents with regards to QA-related aspects. Describe the extent the QA organization reviews and concurs with those Engineering and Administrative Procedures addressed in Table 17.1-1. (6A2)

260.23 Describe in more detail the extent a master list or equivalent document control system is established to identify the current revision of instructions, procedures, specifications, drawings, and procurement documents. When such a list is used, it should be updated and distributed to predetermined responsible personnel. (6B2)

260.24 Describe those provisions which assure that procedures are established to provide for the preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant design. (6C1)

260.25 Describe in more detail the responsibilities of the QA organization for the control of purchased material, equipment, and services, including interfaces between design, procurement, and QA organizations. (7A2)

260.26 Describe those provisions which assure that procurement of spare or replacement parts for structures, systems, and components important to safety is subject to present QA program controls, to codes and standards, and to technical requirements equal to or better than the original technical requirements, or as required to preclude repetition of defects. (7A4)

260.27 Describe those provisions which assure that items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work. (7B2)

260.28 Describe those provisions which assure that suppliers' certificates of conformance are periodically evaluated by audits, independent inspections, or tests to assure they are valid and the results documented. (7B5)

260.29 Describe how correct identification of material, parts, and components is verified and documented prior to release for fabrication, assembling, shipping, and installation. (8B3)

260.30 Describe the responsibilities of the QA/QC organization for the qualification of special processes, equipment, and personnel and in assuring that these qualifications have been satisfactorily performed. (9A2 and 9B1)

260.31 Describe in more detail those measures which assure that procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. (9B2)

260.32 Describe those measures which assure that qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current. (9B3)

260.33 Describe those provisions which assure that an effective inspection program has been established which provides criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required. Describe the responsibilities of the QA/QC organizations in the above functions. (10A)

260.34 Describe in more detail the organization responsible for inspection and provide assurance that individuals performing inspections are other than those who performed or directly supervised the activity being inspected and do not report directly to the immediate supervisors who are responsible for the activity being inspected. If the individuals performing inspections are not part of the QA/QC organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure such as cost and schedule should be reviewed and found acceptable by the QA organization prior to the initiation of the activity. (10B1)

260.35 Describe those provisions which assure that inspection procedures, instructions, or checklists provide for the following as determined by the QA/QC organization: (10C1)

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of item 10B1.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy requirements.

260.36 Describe those provisions which assure that procedures are established and described to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector. (10C2)

260.37 Describe in more detail those provisions which assure that inspection results are documented, evaluated, and their acceptability determined by a responsible individual or group. (10C3)

260.38 Describe the provisions which assure that when inspections associated with normal operations of the plant (such as routine maintenance, surveillance, and tests) are performed by individuals other than those who performed or directly supervised the work, but are within the same group, the following controls are met: (SRP Section 17.2.10, item 2)

a. The quality of the work can be demonstrated through a functional test when the activity involves breaching a pressure retaining item.

b. The qualification criteria for inspection personnel are reviewed and found acceptable by the QA organization prior to initiating the inspection.

260.39 Clarify that test procedures or instructions provide as required for the following: (11B1)

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required)

d. Provisions for assuring test prerequisites have been met.

260.40 Describe in more detail the role of the QA and other organizations responsible for establishing, implementing and assuring effectiveness and consistency of the calibration program. (12.2)

260.41 Describe in more detail those provisions which assure that measuring and test equipment is labeled or tagged or "otherwise controlled" to indicate due date of the next calibration. The method of "otherwise controlled should be described. (12.5)

260.42 Describe those provisons which assure that calibrating standards have a greater accuracy than standards being calibrated. (12.7)

260.43 Describe those provisions which assure that calibration of measuring and test equipment be against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated or, when this is not possible, have an accuracy that assures the equipment being calibrated will be within required tolerance and that the basis of acceptance is documented and authorized by responsible management. The management authorized to perform this function is identified. (12.6)

260.44 Describe those provisions which assure that the QA program provides controls for the storage of chemicals, reagents (including control of shelf life), lubricants, and other consumable materials. (SRP Section 17.2.12, item 2)

260.45 Describe those provisions which assure that procedures are established to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval. (14.3)

260.46 Describe those provisions which assure that QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for independent review and for the disposition of nonconforming items. (15.1 & 15.2)

260.47 Clarify that measures will assure that an audit plan is prepared identifying audits to be performed, their frequencies, and schedules. (18A2)

260.48 Modify Table 17.0-1 to include a commitment to Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (endorses N45.2.4).

260.49 Since the WASH documents 1283 and 1309 have been superceded by Regulatory Guides it is recommended that Section 17 of the FSAR delete any reference to these WASH documents.

260.50 Inconsistencies appear to exist on commitments and alternates to Regulatory Guides described in Chapter 1.8 and 17.0 of the FSAR. It is recommended that Chapter 1.8 describe any alternatives to the Regulatory Guides and that Chapter 17.0 simply reference Chapter 1.8 in regards to commitments and alternatives to Regulatory Guides.

Nine Mile Point 2
ENVIRONMENTAL QUALIFICATION

Request for Additional Information

270.3 Prior to the completion of our review of your license application, it is necessary that we establish that you comply with the Commission's requirements applicable to environmental qualification contained in 10 CFR 50.49 for electrical equipment important to safety; GDC 4, Appendix A, 10 CFR 50; and Appendix B, 10 CFR 50, Sections III, XI, XVII.

As a result of the issuance of Section 50.49 of 10 CFR Part 50, some of the information requested in SRP3.11 and R.G. 1.70, Section 3.11, is no longer required for staff review. Other new information is required, however, and is defined in this guidance. By utilizing these guidelines to demonstrate compliance with the Commission's regulations, applicants can significantly reduce the need for requests for additional information from the NRC staff. The information required may be submitted in Section 3.11 of the FSAR or in a separate submittal. If the latter approach is chosen, Section 3.11 should reference the information in the environmental qualification program submittal. The following guidelines summarize the information to be furnished to the staff:

a. The applicable criteria should be identified and shown to have been incorporated into the environmental qualification program.

b. The systems and components selected for harsh environment qualification should be identified and demonstrated to be complete. Correlation with Table 3.2.1 of the FSAR should be provided for identification of safety-related equipment. Safety-related equipment exempted from harsh environment qualification requirements should be justified.

To demonstrate compliance with 50.49 (b) (2) concerning nonsafety-related electrical equipment whose failure could prevent the satisfactory accomplishment of safety functions, and 50.49 (b) (3), post-accident monitoring equipment, the following information should be provided:

(1) a list of all nonsafety-related electrical equipment, located in a harsh environment, whose failure under postulated environmental conditions could prevent the satisfactory accomplishment of safety functions by the safety-related equipment. A description of the method used to identify this equipment must also be included. The nonsafety-related equipment identified must be included in the environmental qualification program.

(2) a list of all post-accident monitoring equipment currently installed, or that will be installed before plant operation, that is specified as Category 1 and 2 in Revision 2 of R.G. 1.97 and is located in a harsh environment. The equipment identified must be included in the environmental qualification program. In addition, any TMI Action Plan equipment previously committed to installation prior to fuel load should be identified and qualified in accordance with the applicable criteria.

c. The normal, abnormal, and accident environments should be provided for each plant zone. References should be made to other FSAR sections, where appropriate, for methodologies used to determine accident environments. The requirement for calculation of the radiation doses to equipment in close proximity to recirculating fluid systems inside and outside containment for LOCA events in which the primary system does not depressurize should be incorporated into the program (II.B.2 of TMI Action Plan, NUREG-0737). The time dependent environments should be defined for accident conditions.

d. The qualification methodology should be summarized by reference to appropriate criteria (Reg. Guides, industry standards, etc.) and should address the following areas:

(1) Margin

(2) Aging

(3) Dose rate and synergistic effects

(4) Use of analysis for qualification

(5) The maintenance/surveillance program, in particular its conformance with R.G. 1.33 and the industry standard it endorses, and its use in the aging program for equipment qualification.

e. All equipment located in a harsh environment should be identified by its tag number, and its location and operability time provided. Mild environment equipment need not be included in this list. For electrical equipment, the information requrested in Appendix E of NUREG-0588, and SRP 3.11 concerning test results should be submitted. An acceptable format for this information was provided with IE Bulletin 79-01B in the form of "SCEW sheets." Other formats providing the same information may be submitted however.

The information requested in item 4 of Appendix E, NUREG-0588 need not be submitted but should be available for audit by the staff.

f. For mechanical equipment, the staff review will concentrate on materials which are sensitive to environmental effects, for example, seals, gaskets, lubricants, fluids for hydraulic systems, diaphragms, etc. A review and evaluation should be performed that includes the following:

(1) Identification of safety-related mechanical equipment located in harsh environment areas, included required operating time.

(2) Identification of non-metallic subcomponents of this equipment.

(3) Identification of the environmental conditions this equipment must be qualified for. The environments defined in the electrical equipment program are also applicable to mechanical equipment.

(4) Identification of non-metallic material capabilities.

(5) Evaluation of environmental effects.

The list of equipment identified should be submitted. From this list the staff will select approximately three items of mechanical equipment for which documentation of their environmental qualification should be provided for review. Also, the results of the review should be provided for all mechanical equipment in harsh environment areas and corrective actions identified. Justification for interim operation must be submitted prior to fuel load for any mechanical equipment whose qualification cannot be established.

Upon receipt of the above information, we will review your environmental qualification program for compliance with 10 CFR 50.49, and Appendix B, 10 CFR 50, and request any additional information needed to establish its acceptability. We will then perform an audit review of your electrical equipment environmental qualification files and associated installed equipment. Following this audit, an SER supplement will be prepared documenting the results of our review and evaluation. Prior to granting of an operating license, we must be able to conclude that you are in compliance with 10 CFR 50.49 and Appendix B, 10 CFR 50.

Attachment 6

Nine Mile Point 2
Equipment Qualification (Seismic and Dynamic)
Request for Additional Information

The request for additional information described below are in connection with the EQB review of equipment and seismic and dynamic qualification, including operability qualification of mechanical equipment. In many cases, reference has been made to the appropriate portions of the FSAR.

Seismic and Dynamic Qualification

271.3 Provide the date when the tables which give data on seismic qualification of equipment will be completed. (Ref. Tables 3.10A and 3.10B).

271.4 For those cases where a different valve operator is used to qualify a safety-related valve operator (a different valve operator is any operator for which there are deviations in make, model number, design, size, or materials), provide the methodology used to justify testing a different operator. Justifications and similarity comparisons should consider, among other things, the effects of aging (including thermal, seismic and vibration cycles), seismic vibration, and margins used in establishing test response spectrum.

271.5 In selected cases (refer to FSAR pp. 3.10 B-4, 5 and FSAR sections 3.9.2.2.2 14-B and 3.9.3.2.6.B) qualification of safety-related valve operators was to IEEE 382-1972. When completing the summary tables which include equipment qualification information, indicate what criteria are used for qualification of other types of operators such as air or hydraulic.

Operability Qualification of Mechanical Equipment

271.6 Provide justification regarding the use of analysis only in qualifying the ECCS pump motor assembly and the RCIC pump assembly. Clarify why pumps were not included generically in the test list (Ref. 3.9 A-10, Dynamic Testing).

271.7 Excepting those valves where qualification is identified by the parent valve listed, provide the basic criteria used in selecting the representative valve for qualification testing.

271.8 Identify if there are any manually operated valves which are required to change position for any safety system to perform its function; indicate the impact of its failure on safety function. These valves are distinguished from those "Man Signal" valves in Table 3.9 A-12 which presumably are remote manual.

271.9 It does not appear that the HPCS diesel generator is included among the equipment qualified in FSAR section 3.9 A-4. Provide information showing qualification for this unit, and for its appurtenances (such as a transfer pump if a separate day tank for fuel is used) and air starting equipment). (Ref. Table 3.9 A-9)

271.10 Identify if there are any safety-related deep draft pumps in the plant.

Fire Protection
Request for Information
Nine Mile Point Nuclear Station Unit 2

280.2 The fire protection program will be reviewed to the guidelines of BTP CMEB 9.5-1 (NUREG-0800), July 1981. You indicated that a comparison to the guidelines, including identification and justification of deviations, would not be available until the third quarter of 1984. We will need this information by December 1, 1983 to meet your schedule of fuel load.

280.3 Provide the qualifications of the fire protection engineer responsible for the formulation and implementation of the fire protection program.

280.4 Verify that administrative controls will be developed and implemented to comply with BTP CMEB 9.5-1 Section C.2.

280.5 Verify that the plant fire brigade, fire brigade equipment and fire brigade training program are in accordance with BTP CMEB 9.5-1 Section C.3.

280.6 Substantiate the fire resistance capability of the barriers used to separate safety-related areas or high hazard areas by verifying that their construction is in accordance with a particular design and that has been fire tested. Describe the design, the test method used and the acceptance criteria. Provide information for the following components.

(a) Rated fire barriers, including floor and ceiling construction and the support for barriers that are not floors or ceilings;

(b) Fire dampers and fire doors, including a description of how they are installed in the ventilation ducts that penetrate rated fire barriers of safety-related areas; and

(c) Fire barrier penetration seals around cuts, pipes, cables, cable trays and in other openings (e.g. concrete joints sealers and fillers) including verification that all seals are of the thickness specified in the tests, and that cables and cable trays are supported in a manner similar to supporting arrangements used in any tests.

280.7 Verify that the closing of fire doors will be supervised by one of the measures stated in BTP CMEB Section C.5.a.

280.8 Verify redundant safety-related cable systems outside the cable spreading room are protected to comply with BTP CMEB 9.5-1 Section C.5.e(2).

280.9 Verify that fire protection has been provided for safe shutdown so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage and that systems necessary to achieve and maintain cold shutdown from either the control room or the emergency control station(s) can be repaired within 72 hours.

Provide an analysis which shows that one redundant train of equipment structures, systems, and cables necessary for safe shutdown can be maintained free of fire damage by either:

(1) Separation of cables and equipment and associated circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers should be protected to provide fire resistance equivalent to that required of the barrier;

(2) Separation of cables and equipment and associated circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards. In addition, fire detectors and an automatic fire suppression system should be installed in the fire area; or

(3) Enclosure of cable and equipment and associated circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system should be installed in the fire area.

280.10 Identify those areas of the plant that will not meet the guidelines of Section C.5.b of BTP CMEB 9.5-1 and, thus alternative shutdown will be provided. Additionally provide a statement that all other areas of the plant will be in compliance with Section C.5.b of BTP CMEB 9.5-1.

For each of those fire areas of the plant requiring an alternative shutdown system(s) provide a complete set of responses to the following requests for each fire area:

(1) List the system(s) or portions thereof used to provide the shutdown capability with the loss of offsite power.

(2) For those systems identified in (1) for which alternative or dedicated shutdown capability must be provided, list the equipment and components of the normal shutdown system in

the fire area and identify the functions of the circuits of the normal shutdown system in the fire area (power to what equipment, control of what components and instrumentation). Describe the system(s) or portions thereof used to provide the alternative shutdown capability for the fire area and provide a table that lists the equipment and components of the alternative shutdown system for the fire area. For each alternative system, identify the function of the new circuits being provided. Identify the location (fire zone) of the alternative shutdown equipment and/or circuits that bypass the fire area and verify that the alternative shutdown equipment and/or circuits are separated from the fire area in accordance with Section III.G.2.

(3) Provide drawings of the alternative shutdown system(s) that highlight any connections to the normal shutdown systems (P&IDs for piping and components, elementary wiring diagrams of electrical cabling). Show the electrical location of all breakers for power cables, and isolation devices for control and instrumentation circuits for the alternative shutdown systems for that fire area.

(4) Verify that procedures have been or will be developed that describe tasks to be performed to effect the shutdown method. Provide a summary of these procedures outlining operator actions.

(5) Verify that the manpower required to perform the shutdown functions using the procedures of (4) as well as provide fire brigade members to fight the fire is available as required by the fire brigade technical specifications.

(6) Provide a commitment to perform adequate acceptance tests of the alternative shutdown capability. These tests should verify that: equipment operates from the local control station when the transfer or isolation switch is placed in the "local" position and that the equipment cannot be operated from the control room; and that equipment operates from the control but cannot be operated at the local control station when the transfer isolation switch is in the "remote" position.

(7) Verify that repair procedures for cold shutdown systems are developed and material for repairs is maintained on site. Provide a summary of these procedures and a list of the material needed for repairs.

280.11 It is our position that you comply with Section C.5.g(c) of BTP CMEB 9.5-1, in that a fixed emergency lighting system consisting of sealed beam units with individual (8-hour minimum) battery power supplies should be installed in all areas required for safe shutdown operations, including access and egress routes. Specify the foot-candles provided at the floor level of access routes and at operational areas.

280.12 Verify that fixed repeaters installed to permit use of portable radio communication units will be protected from exposure fire damage to comply with BTP CMEB 9.5-1 Section C.5.g.

280.13 Verify that Class A fire detection systems have been provided to comply with BTP CMEB 9.5-1 Section C.6.a to protect all areas of the plant which contain or present an exposure fire hazard to safety related equipment and cables.

280.14 Verify that primary and secondary power supplies for the fire detection systems are provided to comply with Section 2220 of NFPA 72D.

280.15 Provide a plot plan of the site showing all fire protection water mains to include pipe sizes, valves, location of hydrants, and fire pump connections. It is our position that the piping be arranged and valved such that a single break will cause the loss of both fire pumps or shut off all fire protection water to any area of the plant. Also, it is our position that standpipe and hose station be provided throughout the plant in accordance with NFPA 14, including permanent hose stations inside containment to meet Section C.7.a of BTP CMEB 9.5-1.

280.16 Verify that provisions have been made to supply water to standpipe and hose connections for manual fire fighting in areas containing equipment required for safe plant shutdown in the event of a safe shutdown earthquake in accordance with Section C.6.c of BTP CMEB 9.5-1.

280.17 Verify that the fire pumps and their controllers are UL listed and installed in accordance with NFPA 20 requirements. The fire pumps start-up setpoints should be adjusted such that both fire pumps do not start simultaneously (at least a 5 to 10 second delay between pump start-ups is required by NFPA 20).

280.18 Verify that the fire pumps can provide, in accordance with BTP CMEB 9.5-1 Section C.6.b, the largest firewater flow and pressure (based on 500 gpm for manual hose streams plus the largest design demand of any sprinkler or deluge system in a safety related areas as determined in accordance with NFPA 13 or NFPA 15) with the largest fire pump out of service.

280.19 Verify that smoke detectors have been provided in all control room cabinets and consoles in accordance with BTP CMEB 9.5-1 Section C.7.b.

280.20 Verify that the fire protection for the PGCC system is in accordance with NEDO 10466-A Rev. 2.

280.21 Identify all mechanisms by which fire or fire fighting activities may cause the simultaneous failure of redundant or diverse trains that have been considered in the design. Describe the measures taken to preclude such failures. Include consideration of other design basis events (e.g. seismic) simultaneously affecting fire protection system in several areas of the plant.

280.22 On page 9A.3-16 of the FSAR you state that primary fire suppression in the cable spreading room is provided by a total flooding Halon fire extinguishing system. It is our position that the primary fire suppression in the cable spreading room be an automatic water system in conformance with BTP CMEB 9.5-1 Section C.7.c.

280.23 On page 9A.3-3 of the FSAR, you state the safety-related equipment subject to water damage is provided with rain-tight enclosures to ensure availability when required, with respect to potential water damage associated with fire fighting activities. Verify that these enclosures provide adequate protection against both sprinkler water discharge and fire hose water discharge, or that safe plant shutdown is not adversely affected by such damage.

280.24 Verify that floor drains sized to remove expected fire fighting water flow without flooding safety related equipment have been provided in those areas where fixed water fire suppression systems have been installed, in conformance with BTP CMEB 9.5-1 Section C.5.a(14).

280.25 Verify flammable liquids will be stored to comply with BTP CMEB 9.5-1 Section C.5.d(4).

280.26 Verify that the manually actuated smoke venting systems provided for electrical tunnels and the cable routing areas comply with BTP CMEB 9.5-1 Section C.5.f.

280.27 On page 9A.3-9 of the FSAR, you state that fire dampers provided at ventilation ductwork penetrations of fire barriers would close in response to large fires and can be manually reopened by plant personnel for smoke removal. Describe the mechanism by which the dampers are closed and the procedure by which the dampers are reopened, including the locations of these dampers and their accessibility for manual reopening.

280.28 On page 9A.3-10 of the FSAR, you state that the power supply and controls for ventilation systems are generally outside the area served except where it is impractical to do so. Describe the areas where power supply and controls for ventilation systems are inside the area served and the consequences of the loss of these systems during a fire.

280.29 On page 9A.3-10 of the FSAR, you state that, although egress stairways, elevator enclosures and chutes are enclosed, hoistways and other unprotected openings exist. Identify and describe the unprotected openings and fire protection provided.

280.30 Verify that smoke and heat vents installed in various plant areas comply with the provisions of BTP CMEB 9.5-1 Section C.5.f.

280.31 Verify that control and sectionalizing valves in the fire water systems are electrically supervised or administratively controlled to comply with the provisions of BTP CMEB 9.5-1 Section C.6.c(2).

280.32 Verify that Halon and carbon dioxide suppression systems comply with the provisions of BTP CMEB 9.5-1 Section C.6.d and C.6.e.

280.33 Verify that the protection of safety-related battery rooms complies with the provisions of BTP CMEB 9.5-1 Section C.7.g.

Additional Information Required

Chemical Engineering

Nine Mile Point Nuclear Station, Unit 2

281-3 (5.4.8) (10.4.6) Verify that the initial total capacity of new demineralizer resins (condensate and primary coolant) will be measured and describe the method to be used for this measurement (Regulatory Position C.3 of Regulatory Guide 1.56, revision 1).

281-4 (10.4.6) Establish and state the sequential resin replacement frequency in order to maintain adequate capacity margin in the condensate treatment system (Regulatory Position C.2 of Regulatory Guide 1.56, revision 1). Include the basis for the resin replacement frequency.

281-5 (10.4.6) Describe the method of determining the condition of the demineralizer units (see p. 10.4.25 of FSAR) so that the ion exchange resin can be replaced before an unacceptable level of depletion is reached (Regulatory Position C.4 of Regulatory Guide 1.56, revision 1). Describe the method by which (a) the conductivity meter readings for the condensate cleanup system will be calibrated, (b) the quantity of the principal ions likely to cause demineralizer breakthrough will be calculated, (c) the flow rates through each demineralizer will be measured, and (d) the accuracy of the calculation of resin capacity will be checked.

281-6 (5.4.8) (10.4.6) Indicate the control room alarm set points of the conductivity meters at the inlet and outlet demineralizers in the condensate and reactor water cleanup systems when either (Regulatory Position C.5 of Regulatory Guide 1.56, revision 1):

a. The conductivity indicates marginal performance of the demineralizer system;

b. The conductivity indicates noticeable breakthrough of one or more demineralizers.

281-7 (5.4.8) (10.4.6) Indicate the reactor coolant limits and corrective action to be taken if the conductivity, pH, or chloride content, as established in the Technical Specifications, is exceeded. Describe the chemical analysis methods to be used for the determination of these values. (Regulatory Position C.6 of Regulatory Guide 1.56, revision 1).

281-8 (10.4.6) Describe the water chemistry control program to assure maintenance of condensate demineralizer influent and effluent conductivity within the limits of Table 2 of Regulatory Guide 1.56, revision 1. Include conductivity meter alarm set points and the corrective action to be taken if the limits of Table 2 are exceeded.

281-9 (10.4.6) In accordance with Regulatory Position C.1 of Regulatory Guide 1.56 revision 1, describe the sampling frequency, chemical analyses, and established limits for purified condensate dissolved and suspended solids that will be performed and the basis for these limits.

281-10 (10.4.6) Tests by EPRI have shown that intergrannular stress corrosion cracking (IGSCC) can be inhibited by keeping the level of impurities in the primary coolant low and the oxygen concentration around 20 ppb. Describe how you will keep the concentration of impurities and of oxygen to a level below where IGSCC is initiated. Describe if you have any plans for oxygen control by hydrogen addition.

281-11 (6.1.2) We are concerned that insoluble debris formed, under DBA conditions inside containment, from unqualified organic paint surfaces and from corrosion products (from galvanized steel and zinc paints without qualified organic top coats) may adversely affect the performance of the RHR or the Containment Spray systems which take suction from the suppression pool.

a. In the FSAR (p. 6.1-5 and Table 6.1-3) you indicate that the total amount of unqualified paint inside containment on NSSS supplied components is minimal. Provide the total amount (area and film thickness) of unqualified paint inside containment on both NSSS and non-NSSS supplied components. We will use this information to estimate the amount of solid debris that can be formed from the unqualified paint under DBA conditions; and can potentially reach the suppression pool. Confirm that qualified paints have been utilized to the maximum extent practicable.

b. Provide an analysis that demonstrates that the insoluble debris from unqualified organic paints and corrosion products from galvanized steel and zinc paints without qualified organic top coat will not cause a sump debris problem or adversely affect the performance of the RHR or containment spray systems.

281-12 (9.1.3) Regarding the Spent Fuel Pool Cleanup System, provide the following information:

Describe the samples and instrumentation and the frequency of the measurements that will be performed to monitor (a) the spent fuel pool water purity and (b) the need for ion exchanger resin and filter replacement. State the chemical and radio-

chemical limits to be used in monitoring the spent fuel pool water and for initiating corrective action. Provide the basis for establishing these limits. Your response should consider variables such as: gross gamma and iodine activity, demineralizer and/or filter differential, pressure, decontamination factor, pH and crud level.

281-13 (NUREG-0737 II. B.3) Provide information that satisfies each of the attached evaluation (attachment 8A) criteria guidelines for post accident sampling system.

ATTACHMENT NO. 8A TO
POST ACCIDENT SAMPLING SYSTEM
NUREG-0737, II.B.3 EVALUATION
CRITERIA GUIDELINES

The post accident samoling system will be evaluated for compliance with the criteria from NUREG-0737, II.B.3. These eleven items have been copied verbatim from NUREG-0737. The licensees submittal should include information equivalent to that which is normally provided in an FSAR. System schematics with sufficient information to verify flow paths should be included, consistent with documentation requirements in NUREG-0737, with appropriate discussion so that the reviewer can determine whether the criteria have been met. Further information pertaining to the specific clarifications of NUREG-0737, which will be considered in the reviewers evaluation are listed below. Technically justified alternatives to these criteria will be considered.

Criterion: (1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Clarification: Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of off-site power (i.e. designate an alternative backup power source, not necessarily the vital (Class IE) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

(a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and non-volatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Clarification: 2 (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation to minimize personnel radiation exposures (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

2 (b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

2 (c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

2 (d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Criterion: (3) Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

Clarification: System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Criterion: (4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification: Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is <0.1 ppm by measurement of a dissolved hydrogen residual of

≥ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Criterion: (5) The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification: BWR's on sea or brackish water sites, and plants which use sea or brackfish water in essential heat exchangers (e.g. shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform a chlorida analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ppm Cl (the licensee should establish this value; the number in tha blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Criterion: (6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).

Clarification: Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Criterion: (7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

Clarification: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Criterion: (8) If inline monitoring in used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samplies. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Criterion: (9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1μ Ci/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification: (9) (a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(9) (b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

Criterion: (10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Clarification: The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

In general this analysis should be accurate within ±5% of the measured value (i.e. at 6,000 ppm B the tolerance is ± 300 ppm while at 1,000 ppm B the tolerance is ± 50ppm). For concentrations below 1,000 ppm the tolerance band should remain at ± 50 ppm.

- Chloride: measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

- Hydrogen or Total Gas: monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of ± 10% is desirable between 50 and 2000 cc/kg but ± 20% can be acceptable. For concentration below 50 cc/kg the tolerance remains at ± 5.0 cc/kg.

- Oxygen: monitored to assess coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

- pH: measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within ±0.3 pH units. For all other ranges ± 0.5 pH units is acceptable.

To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituent	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs^+	250	Cesium Nitrate
Ba^{+2}	10	Barium Nitrate
La^{+3}	5	Lanthanum Chloride
Ce^{+4}	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li^+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^-	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

Criterion: (11) In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(11)(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

SITING ANALYSIS
REQUEST FOR ADDITIONAL INFORMATION
ON NINE MILE POINT UNIT 2

311.6 (2.1.3) There seems to be some discrepancy in the population distribution figures when comparing the Table 2.1 series versus the Table 2L series. Explain the differences and indicate which series are correct. Revise the tables if warranted.

311.7 Please provide information on any schools, churches, recreational facilities, prisons, nursing/convalescent homes, day care centers, hospitals, or other institutions within 10 miles of the site. Designate their locations i.e., distance/direction relative to the plant, the capacity of each facility, and the number of persons normally in attendance, employed or housed, etc.

311.8 (2.1.3.4) Section 2.1.3.4 of the FSAR states that the Ontario Bible Campground at Lakeview is located approximately 2.4 miles WSW of the station. Table 2.1-1 of the ER indicates that Lake View is 1.0 mile SW of the station. In the PSAR, however, it is listed as being about 4500 ft. SW of Unit 1. Please clarify and update your documentation for consistency.

311.9 (2.1.3.5) The population of Oswego has been steadily decreasing; -5.6% (22,155 to 20,913) from 1960 to 1970, and -5.4% (20,913 to 19,793) from 1970 to 1980. Considering this decline, on what basis do you project that the population will increase at least 26.3% (25,000) by 1990?

REACTOR SYSTEMS
NINE MILE POINT NUCLEAR STATION UNIT 2

6.3 EMERGENCY CORE COOLING SYSTEMS

Request for Additional Information

440.30 (6.3) FSAR Subsection 6.3.1.1.2 states the following:

"Systems that interface with, but are not part of the ECCS are designed and operated in such a way that failure(s) in the interfacing systems do not propagate to and/or affect the performance of the ECCS".

(1) Identify those systems which interface with the ECC systems.

(2) Demonstrate in detail how you achieve adequate ECCS performance with failures in the interface systems.

440.31 (6.3) Figures 6.3-3, 6.3-4 and 6.3-5 do not provide required and available NPSH for ECCS pumps in all modes of operation.

Verify that adequate NPSH margin exists for ECCS pumps in all modes of operation. Provide detailed NPSH calculations to demonstrate conformance to Regulatory Guide 1.1 for the ECCS pumps. Revise figures 6.3-3, 6.3-4 and 6.3-5 to show both required and available NPSH.

440.32 (6.3) Tables 7.3-3 and 7.6-1 provide differential pressure instrument range for LPCS injection valve.

(1) Specify the pressure interlock setpoint of the LPCS isolation valve (F005).

(2) Demonstrate that the LPCS pump developed head plus ΔP setpoint for the pressure interlock at the isolation valve would not exceed the design pressure of the LPCS piping system.

440.33 (6.3) Verify the recirculation flow control valve closure rate during the most limiting LOCA case that would not violate the 10 CFR 50.46 PCT limit. Also discuss the provisions made for the recirculation flow control valve closure rate.

440.34 (6.3) Verify that all ECCS suction lines are adequately submerged to preclude formation of an undesirable vortex formation

440.35 (6.3) When the water level in the condensate storage tank drops to a predetermined level, the HPCS pump switches automatically to the suppression pool.

Provide assurance that adequate NPSH exists up to switchover.

440.36 (6.3) Provide detailed calculations which demonstrate that the NPSH available at all points in ECCS pump suction piping

is adequate to preclude local flashing and pump cavitation under the worst postulated conditions.

440.37 (6.3) Discuss the adequacy of design provisions to mitigate possible water hammer in pressure relief valve discharge lines for ECC systems.

440.38 (6.3) The ECCS contains manual as well as motor-operated valves. Consideration must be given to the possibility that manual valves might be left in the wrong position and remain undetected when an accident occurs. Discuss the provisions made for each manual valve to minimize the possibility of such an occurrence.

440.39 (6.3) Discuss the design provisions that are incorporated to facilitate maintenace and continuous operation of the ECCS pumps, seals, valves, heat excahngers and pipings in the long-term LOCA mode of operation considering that the water being recirculated is potentially very radioactive.

440.40 (6.3) Response to the following TMI action items are required to complete the review of section 6.3.

(a) <u>II.K.1 IE bulletins on measures to mitigate small break LOCA's and Loss-of Feedwater Accidents.</u>

(i) <u>II.K.1.5 Review of ESF Valves</u>

FSAR section 1.10 does not address this item. The staff position is given below:

NRC Position

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Confirm that Nine Mile Point Unit 2 performed or will perform the review described above.

(ii) II.K.1.10 Operability Status

FSAR section 1.10 does not address this item. The staff position is given below:

NRC Position

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Confirm that Nine Mile Point Unit 2 performed or will perform the review described above.

(b) II.K.3 Final Recommendations of Bulletins and Orders Task Force

(i) II.K.3.17 We require a commit ment from the applicant that they report ECCS outages via LERs and report summary of outages via annual reports.

(ii) II.K.3.18 Modification of Automatic Depressurization System Logic.

FSAR section 1.10 does not address this item. The staff completed its evaluation of the BWR Owners Group generic response to Item II.K.3.18. Provide the plant specific response based on the staff evaluation report.

METEROLOGY and EFFLUENT TREATMENT REQUEST FOR ADDITIONAL INFORMATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

F451.8
(2.3.1.2.2)
SRP 2.3.1
I.3

Regional Climatology

Present the fastest-mile windspeed extremes historically recorded in the vicinity of the site. Both Oswego, New York (WBO) and Rochester, New York (WBO) have been known to exceed the 69 mph value stated in FSAR Section 2.3.1.2.2*.

F451.9
(2.3.2.1)
SRP 2.3.2
II.2

Local Meteorology

Describe the methodology used to substitute meteorological data, other than wind direction, at the three tower levels to improve data recovery for the meteorological parameters shown in Table 2B-2. The staff calculated that the five year joint data recovery for the 61(m) level wind measurement and 61-8(m) delta temperature was 83.8%, using unmodified data from the magnetic tape provided to the staff. Present the actual data recovery for the onsite meteorological parameters used in the dispersion calculations. Explain the reason for long periods of missing data (> 120% consecutive hours).

F451.10
(2.3.3.2)
SRP 2.3.3

Onsite Meteorological Measurements Program

The analog and digital system accuracies and the sensor for temperature difference measurements in Table 2.3-5A do not meet the

*Changery, M., "Historical Extreme Winds for the United States, Great Lakes and Adjacent Regions," National Climatic Center, NOAA, August 1982.

recommended accuracies in Regulatory Guide 1.23. However, it is stated in FSAR Section 2.3.3.2.2 (Amendment 3, June 1983) that Regulatory Guide 1.23 recommended accuracies are satisfied. Clarify this apparent discrepancy.

F451.11
(2.3.3)
SRP 2.3.3

Operational Onsite Meteorological Measurements Program

Meteorological instrumentation, including siting of sensors, sensor performance specifications, the quality assurance program for sensors, and recorders, have not been presented for the back-up and supplemental meteorological measurements program.

1) Present a detailed map showing the relationship of Unit 2 to each of the meteorological monitoring stations. Figures 2.1-1 and 2.3-40 do not identify the location of the J. A. Fitzpatrick back-up tower or the inland meteorological tower.

2) Demonstrate with at least one coincident annual cycle of data from the primary tower level that measurements made on the backup tower are representative of the general site area.

3) Discuss the application of the J. A. Fitzpatrick back-up tower parameter display in the control room at NMP-2.

F451.12
(2.3.4.3)
SRP 2.3.4

Short-Term Dispersion Estimates

The transition from an unstable air mass over the lake to a stable air mass over the land occurs most often in fall and early winter, when the lake is still warm. How is this accounted for in the short-term diffusion model estimates?

F451.13
(2.3.5)
SRP 2.3.5
III.3

Long-Term Dispersion Estimates (Windspeed)

Windspeeds measured at the 61 meter level of the Nine Mile Point meteorological tower are adjusted to the stack, vent or 10(m) release height using a power law relationship (FSAR Section 2.3.5.3.6).

1) Extrapolating windspeeds from 61(m) to higher elevations 130(m) can lead to erroneous results when compared to actual measured values. Justify the use of the power law extrapolation of wind speed in this type of coastal environment.

   a) Include in your discussion the estimated error incured when applying the 10-61(m) layer stability to a much larger (10(m)-130(m)) layer for use in the power law extrapolation.

2) Demonstrate that, for very stable conditions, the power law coefficient (q=0.3) in equation 2.3-20 is more appropriate than other coefficients (q=0.55 or 0.60) recommended in other literature*. What conservative approximations have been made to prevent over estimation of windspeeds at the stack, vent and 10(m) level?

F451.14
(2.3.5.3)
SRP 2.3.5
III.3

Long-Term Dispersion Estimates (Wind Direction)

The wind direction at the 61 meter level is used to represent conditions at the stack release height 130(m). Provide an analysis

*Hanna, S. R., et. al., Handbook on Atmospheric Diffusion, Atmospheric Turbulence and Diffusion Laboratory, NOAA, 1982.

and/or data that show wind direction at the 61(m) level is representative of the stack release height of 130(m). How is a wind direction reversal aloft, common in the spring and early summer with the onset of a lake breeze, accounted for in the dispersion modeling of both routine and non-routine stack release?

F451.15
(2.3.5.3.9)
SRP 2.3.5
(R.G. 1.111)

Land/Lake Breeze Influence on Dispersion

Modifications to the straight-line Gaussian dispersion model (EQ 2.3-19) are necessary when the meteorological data available is unable to account for airflow characteristics of the site.

1) Provide estimates of seasonal (spring and summer) frequencies of lake breeze conditions at the NMP, Unit 2, site using onsite meteorological data *. Present the criteria used to identify the onset of a lake breeze front.

2) Provide estimates of seasonal (spring, summer, fall, and winter) formations and frequency of turbulent internal boundary layers (TIBL's). Theoretical predictions of characteristics of the TIBL are described by W. Lyons and several other authors in "Critical Review of Studies on Atmospheric Dispersion on Coastal Regions," NUREG/CR-2754.

3) Provide an estimate of the seasonal frequency of plume interception with the TIBL for routine and non-routine

*Niagara Mohawk Inland Supplementary Meteorological Tower Study, 1982.

elevated releases from the primary vent and stack, and the basis for these estimates.

4) Spatial and temporal variations in airflow trajectories, particularly airflow reversals during the onset of the lake breeze and curved trajectories during the decay of the lake breeze, have not been explicitly incorporated into the annual average transport and diffusion model for the NMP Unit No. 2 site. Recent comparisons of the results of variable-trajectory models with the results of the straight-line model at coastal nuclear plants (e.g., Perry and St. Lucie) have indicated that the straight-line model may underpredict X/Q values by factors of two to four. Provide further justification for not modifying the results of the straight-line model to consider spatial and temporal variations in airflow such as would be experienced during the onset and decay of the lake breeze.

460.5 (11.2) In Subsection 11.2.2.1 and 11.2.2.2 of the FSAR, you state that overflows from the waste and drain collector surge tanks are directed to the respective floor compartments which are sloped to permit the use of a portable and submersible pump to remove the overflows. Describe in detail (1) interlocks provided for the waste inflows into these tanks with the waste tank levels to prevent any overflows, (2) the provisions provided for waste

high level alarm annunciation in the radwaste and the main control room, and (3) the means of routing the tank overflows to the liquid radwaste treatment system. Provide a basis for justification that your proposed method of handling the tank overflows meet the intent of the Regulatory Position in Regulatory Guide 1.143, Rev. 1, dated October 1979.

460.6 (11.2) In Subsections 11.2.2.1 and 11.2.2.6 of the FSAR, you state that the condensate regeneration subsystem will be used for regeneration and ultrasonic cleaning of the radwaste demineralizer resins. State if the condensate regeneration subsystem is designed to handle, clean, and regenerate the radwaste demineralizer resins in light of the fact that radwaste demineralizers use different volumes of resins (anion and cation resins).

460.7 (11.2) State the expected percents of the processed liquid radwaste to be discharged to the environment from each liquid radwaste subsystem.

460.8 (11.2) In Subsection 11.2.3.1 of the FSAR, you state that Figure 11.5-8 shows all systems that feed the discharge bay and Figure 9.2-14 shows the physical location of the discharge into the bay. These figures show other subjects. Provide the correct figure numbers or new figures as stated in the subsection.

460.9 (11.2) Explain the discrepancy for the amount of average daily floor drains in Table 11.2-3 of the FSAR (8,800 gallons per day) and in Table 3.5-1 of the ER (11,200 gallons per day).

460.10 (11.2) State if NMP-2 has an onsite laundry facility. If so, describe treatment provided for detergent waste prior to discharge to the environment.

460.11 (11.2) Provide a listing of outdoor tanks which may contain potentially radioactive liquid complete with the tank level monitoring provisions.

Indicate whether the outdoor tanks have a dike or retention pond capable of preventing run-off in the event of a tank overflow, and have provisions for sampling and for processing collected liquid radwaste in the liquid radwaste systems.

460.12 (11.3) In Subsection 11.3.1(3) of the FSAR, you state that the offgas system is designed to withstand the effects of a hydrogen detonation. State detonation design pressure and provide an analysis of the design basis for detonation pressure.

460.13 (11.3) In Subsection 11.3.2.1 of the FSAR, you describe three (3) hydrogen analyzers to monitor hydrogen concentrations in the offgas process flow lines:

(a) indicate whether each hydrogen analyzer annunciate locally, as well as in the control room;

(b) indicate whether all hydrogen analyzers are nonsparking and whether these analyzers are able to withstand a hydrogen explosion;

(c) show these analyzers with their controls, alarms, etc., on the gaseous radwaste piping and instrument diagrams; and

(d) describe automatic control features to reduce potential for explosion at high alarm setpoint.

460.14 (11.3) In Subsection 11.3.2.1 of the FSAR, you describe control features of the offgas system. State the location of the control panel for the offgas system and its alarm provision provided in the main control room.

460.15 (11.4) State any exceptions and/or deviations from the Werner & Pfleiderer Corporation Topical Report No. WPC-VRS-001, Rev. 1, dated May 1978, in design, installation, and operation of the NMP-2 solid radwaste treatment system.

460.16 (11.4) Identify the NMP-2 interfaces with the system described in Werner & Pfleiderer Topical Report No. WPC-VRS-001, Rev. 1.

460.17 (11.4) Provide the plant fire hazards analysis information in accordance with CMEB Branch Technical Position 9.5-1, Rev. 2, dated July 1981, to show how safety related systems will be protected from possible fire associated with the combustibles contained in the solid radwaste system.

460.18 (11.4) Provide the NMP-2 Process Control Program establishing a set of process parameters and boundary conditions within which reasonable assurance can be given that solidification will be complete with essentially zero-free liquid based on the recommended WPC-VRS Topical Report process parameters. Your process control program should include administrative and/or plant operating procedures to meet waste form structural stability requirements set forth in Section 61.56 of 10 CFR Part 61.

460.19 (11.4) Section 20.311 of 10 CFR Part 20 requires that any licensee who transfers radioactive waste to a land disposal facility must classify the waste according to Section 61.55 of 10 CFR Part 61. Provide a compliance program to assure proper classification of waste in accordance with guidance provided in Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification, Rev. 0, dated May 1983.

460.20 (11.5) Figure 11.5-7 and Subsection 11.3.2.1 of the FSAR shows and states that there are two (2) radiation monitors with automatic control features (offgas discharge isolation) while Figure 11.3-1b shows only one offgas radiation monitor without automatic control features. Rectify the discrepancies. Incorporate two radiation monitors (RE 104 and 105), shown in Figure 11.3-1c, into Table 11.5-1 of the FSAR.

460.21 (11.5) Regulatory Guides 4.15 (Rev. 1) and 1.143 (Rev. 1) provide guidance for quality assurance for continuous effluent monitoring systems and for radwaste management systems. Indicate whether your process and effluent radiation monitoring system design meets the criteria of these regulatory guides.

460.22 (6.5 and 9.4) In Table 1.8-1 of the FSAR, you have proposed alternative approaches (methods) to the Regulatory Positions in Regulatory Guides 1.52 and 1.140. Provide the additional information to complete our evaluation for acceptance on your proposed alternative approaches as delineated in the following subparagraphs:

(a) Regulatory Guide 1.52, Regulatory Position C.2.a. You state that a demister will be provided only where entrained water droplets could be present. State which ESF system is not provided with a demister and provide your detailed analysis and justification for not providing a demister.

(b) Regulatory Guide 1.52, Regulatory Position C.2.h. You state that all instrumentation and equipment controls are not designed to IEEE-279, 1971. State which instrumentation and equipment controls are not designed to IEEE-279, 1971.

(c) (See question 640.05.1)

(d) Regulatory Guide 1.52, Regulatory Position C.3.d. You state that all HEPA filters will be shipped to an NRC Quality Assurance Station for testing. The NRC does not have quality assurance stations and all HEPA filters should be tested by the manufacturer prior to installation.

(e) Regulatory Guide 1.52, Regulatory Position C.3.h. You state that you have taken an exception relative to the requirements of drain sizes and arrangement, and instead you provided water seals and traps. Provide a typical isometric drawing showing normally open manual valve, water seals and traps, including their respective sizes.

(f) Regulatory Guide 1.52, Regulatory Position C.3.l. You state that fan inlet and outlet pressure losses will not be calculated in accordance with AMCA 201. An estimated resistance for fan inlet and outlet is acceptable and you should document it accordingly.

(g) Regulatory Guide 1.52, Regulatory Position C.3.n. You state that you may not have a resonant frequency above 25 Hz for the unsupported plate or sheet sections. Provide an expected resonant frequency for the unsupported plant and sheet sections.

(h) Regulatory Guide 1.52, Regulatory Position C.4.d. You state that you may reconsider the requirement for ESF system run test (10 hours per month). Provide your justification with criteria for the amount of moisture content on the filters.

(i) Regulatory Guide 1.52, Regulatory Position C.2.f. Same question as (c) above.

(j) Regulatory Guide 1.52, Regulatory Position C.3.i. Same question as (f) above.

460.23 (10.4) For the main condenser evacuation system (MCES), indicate whether the steam air ejectors and associate air ejector lines are designed to withstand the effects of an explosion, and indicate what design provisions in the MCES are made to stop continuous leakage paths after an explosion.

460.24 (15.7) In your analysis of postulate radioactive releases due to a waste gas system leakage failure, you have used exposure limit parameters of less than 5 rem whole body and less than 30 rem for beta. Branch Technical Position 11-5, Rev. 0, 1981, in NUREG-0800 uses 2.5 rem whole body for the system designed to withstand explosions and earthquakes. Your criteria for exposure limit should therefore be 2.5 rem instead of 5 rem. Otherwise, provide your justification for use of 5 rem total body criteria for our evaluation.

CONTAINMENT SYSTEMS
REQUEST FOR ADDITIONAL INFORMATION FOR
NINE MILE POINT UNIT 2

480.01 (6.2.1) FSAR Section 6.2.1.1.2 provides an analysis which determined that the design containment negative differential pressure would not be exceeded. Provide a discussion of the spectrum of depressurization transients which you analyzed to determine the limiting case. List the transients considered as well as the assumptions used and the resulting maximum differential pressures. Indicate the ultimate negative differential pressure capability of the containment

480.02 (6.2.1) Appendix I to SRP Section 6.2.1.1.C, "Steam Bypass for Mark III Containments" states that the Mark II wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell. The FSAR indicates, however, that the containment spray is manually initiated. The staff has allowed operator action only if spray initiation is not required for at least thirty minutes into the postulated LOCA. Justify the use of the manual containment sprays in light of the SRP acceptance criteria. Verify that they are classified as an engineered safety feature. In addition, provide the following information concerning the steam bypass calculation for the small break case: the assumptions concerning plant shutdown time, spray initiation time and rate, and static heat sink assumptions used in the containment pressure anaysis.

Show the containment pressure response following operator action. Also, as a function of time, provide the heat removal percentage contribution due to each heat sink (i.e., spray and static heat sink).

480.03 (6.2.1) Provide the maximum calculated, the design, and the ultimate differential pressure capability for the drywell floor for both upward and downward loads.

480.04 (6.2.1) Attachment 1.9-42 of Table 1.9-1 of the FSAR indicates that the issue of visual inspection of the drywell for possible leak paths will be addressed in the Technical Specifications. While the exact wording in the Tech Specs is unnecessary at this time, the staff does need information concerning NMPC's intent to meet this provision of Appendix I of SRP 6.2.1.1.C. State your intentions in this regard. Similarly, discuss your intentions concerning the vacuum relief valve monthly operability tests as also discussed in SRP 6.2.1.1.C, Appendix I.

480.05 (6.2.1) Attachment 1.9-42 of FSAR Table 1.9-1 indicates that redundant position indicators and alarms are not provided in the drywell-wetwell vacuum breakers because three switches are currently provided on each valve disc. Clarify the existing arrangement of switches to indicate what purpose is served by the three switches present, i.e., discuss the function of each switch. Indicate if redundancy of position indication is provided with the existing arrangement as

well as alarm capability in the control room. If not, justify why a deviation from the SRP should be allowed.

480.06 (6.2.1) Generic Technical Activity A-39, "Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment" was established to resolve, among other things, the concern about steam condensation behavior for Mark II containments. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments" presents the resolution of this issue by indicating: (1) a staff approved acceptance criteria related to suppression pool temperature limits; (2) events for which suppression pool temperature response analyses are required; (3) assumptions to be used in the analysis; and (4) requirements for the suppression pool temperature monitoring system. The acceptance criteria, for instance, state that the suppression pool temperature should not exceed 200°F for all plant transients involving SRV operation during which the steam flux through the quencher perforations exceeds 94 lbm/ft -sec. The temperature limits for other values of steam flux are given in Section 5.1 of the NUREG report.Provide analyses that show the extent to which Nine Mile Point 2 meets the provision of Section 5.1 through 5.7 of the NUREG report including the acceptance criteria, events required to be analyzed, assumptions used, and requirements for suppression pool temperature monitor ing system.

480.07 (6.2.1) NUREG-0802 "Safety/Relief Valve Quencher Loads" includes Appendix A, "Proposed NRC Acceptancce Criteria for Mark II SRV-Related Pool Dynamic Loads". The acceptance criteria are divided into two areas: (1) SRV boundary loads, and (2) T-Quencher tie-down loads. FSAR Section 6A.3 of Appendix 6A, "SRV Loads" addresses the methodology used by NMPC for the evaluation of the SRV discharge system at Unit 2. Many of the specific acceptance criteria identified in Appendix A of NUREG-0801 are not addressed in FSAR Appendix 6A, Section 6A.3. For instance criteria A.1.5, "Suppression Pool Surface Area", on page A-2 indicates that the acceptance criteria are applicable only for plants with a suppression pool area per quencher not less than 204 ft^2. No mention of this criteria is made in the Appendix 6A, Section 6A.3. To facilitate the review of this topic provide a table which indicates the extent of Nine Mile Point 2 complies with each of the specific criteria listed on page A-1 thru A-5 of NUREG-0802. Where the acceptance criteria is not met, justify the deviation.

480.08 (6.2.1) Compare the KWU T-Quencher and support system provided at Nine Mile Point 2 with the KWU quencher and support assembly approved in NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments", Generic Technical Activity A-39. Identify all variations to the one used in NUREG-0802 and provide a sketch comparing any differences between the two systems.

480.09 (6.2.1) SRP Section 6.2.1.2 indicates that the initial atmospheric conditions within a subcompartment should be selected to maximize the resultant differential pressure. The SRP points out that zero percent relative humidity is an acceptable model for air at the initiation of the postulated accident. FSAR Tables 6.2-29," 24-inch Recirculation Suction Break RPV-BSW Annulus" and 6.2-38, the "12-inch Feedwater Line Break RPV-BSW Annulus", indicate that 20 percent, relative humidity was used instead of zero percent. Justify this deviation and indicate what the design margin would have been had zero percent humidity been used.

480.10 (6.2.1) Label the Ordinate Axis of FSAR Figure 6.2-31.

480.11 (6.2.1) FSAR Table 6.2-47 indicates that the design peak pressure difference for the drywell head subcompartment based on analysis of the 24-inch recirculation suction line break will be provided in a future amendment. Indicate when this information will be made available to the staff and show that the design margin will be adequate based on currently available data.

480.12 (6.2.1) Discuss the extent to which node sensitivity studies were performed on the subcompartment analyses to determine the accuracy of the nodalization scheme in regards to the determination of differential pressure for the biological shield wall annulus and drywell head.

480.13 (6.2.1) The FSAR indicates that a high and low pressure drywell bypass leakage test will be conducted. No mention is made of the frequency of testing. Indicate the testing frequency for the low pressure test. (SRP Section 6.2.1.1.c states that the low pressure tests should be performed at each refueling outage). Indicate also if the drywell bypass low pressure leak test flow of 191.7 cfm listed in Table 6.2-61 is 10% of the leakage resulting from a steam bypass capacity of $A\sqrt{K}$ equal to 0.59 ft^2.

480.14 (6.2.1) FSAR page 6.2-43 indicates that the RHR suction strainers prevent the passage of particles larger than 1/4-inch long and 3/32-inch in diameter. Verify that this size spacing is adequate to prevent particles of debris from blocking any part of the flow path of coolant, e.g., the fuel channels, by providing a comparison of the suction strainer opening with the passageways through which coolant flows such as containment spray nozzles, ECCS spargers and fuel channel spacings. Provide a sketch of the suction strainer assembly.

480.15 (6.2.2) Concerning the containment spray system, discuss the effect of downcomers on spray performance and include in your discussion the effect of steaming around hot downcomer vents.

480.16 (6.2.2) Provide a list of all loads used in the design of the RHR intake strainers and provide information that demon-

strates the capability of the strainers to accommodate the hydrodynamic loads from downcomer discharges.

480.17 (6.2.2) FSAR Attachment 1.9-42 indicates that the calculated suppression chamber atmosphere temperature from steam bypass exceeds the design valve of 212°F. Demonstrate that the containment structure especially, the steel liner and the components within the suppression pool have design temperatures which are above the actual temperaures they obtain due to LOCA and steam bypass. Indicate the maximum temperatures achieved by these structures and components.

480.18 (6.2.2) Provide a figure which depicts the relative locations of the RHR suction and return lines in the suppression pool. Discuss the adequacy of this configuration to provide mixing of the return water with total pool inventory.

480.19 (6.2.2) Provide a table of the openings into the secondary containment such as the personnel doors and indicate in the table whether they are:

1. Under administrative control; and
2. Provided with position indication and alarms in the control room.

480.20 (6.2.3) Provide a table listing the high energy lines which pass through the secondary containment and indicate which ones, if any, are covered with guard pipes. For those high energy lines which do not have guard pipes provide the results of analyses to demonstrate that the primary and secondary containment structure are capable of withstanding the effects of a high energy pipe rupture occurring inside the secondary containment without loss of integrity.

480.21 (6.2.3) FSAR Attachment 1.9-46 of Table 1.9-1 "SRP Conformance to Acceptance Criteria" indicates that no analysis was performed for the pressure/temperature response of the reactor building to postulated LOCA. Provide a schedule for the submittal of this information. In performing this analysis consideration should be given to the parameters discussed in SRP Section 6.2.3.II.1. For example, the analysis should include heat transfer from the primary to secondary containment as well as heat transfer from the primary containment atmosphere to the primary containment structure. Also, heat loads generated within the secondary containment should be considered. Provide the details of the analysis and assumptions including the above considerations, used as outlines in SRP 6.2.3.

480.22 (6.2.3) Bypass leakage is defined as that leakage from the primary containment which can circumvent the secondary containment boundary and escape directly to the environ-

ment, i.e., bypassing the leakage collection and filtration system of the secondary containment. FSAR Table 6.5-56 indicates that most piping lines are not potential bypass path. List the lines so designated and indicate why they are not bypass leakage paths. Systems lines may be excluded from consideration as potential bypass paths for reasons such as: the lines terminate in the secondary containment, an air or water sealing system is provided to process or eliminate leakage, or a closed system is proposed for the leakage boundary. If a closed system is proposed as the leakage boundary to preclude bypass leakage verify that the following provisions of SRP 6.2.3 are satisfied. The system should:

a. Either (1) not directly communicate with the containment atmosphere, or (2) not directly communicate with the environment, following a loss-of-coolant accident.

b. Be designed in accordance with Quality Group B standards, as defined by Regulatory Guide 1.26 (Systems designed to Quality Group C or D standards that qualify as closed systems to preclude bypass leakage will be considered on a case-by-case basis.)

c. Meet seismic Category I design requirements.

d. Be designed to at least the primary containment pressure and temperature design conditions.

e. Be designed for protection against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features.

f. Be tested for leakage, unless it can be shown that during normal plant operations the system integrity is maintained.

Specify the estimated bypass leakage for penetrations which must be considered as bypass paths.

480.23 (6.2.3) Provide the following additional information related to potential bypass leakage paths given in Table 6.2-56.

A) For each air or water seal demonstrate that a sufficient inventory of the fluid is available to maintain the seal 30 days following onset of LOCA. Note that the suppression cannot be considered a water seal. Describe the testing and proposed entries for the Tech Specs that will verify the assumptions used in the analyses.

B) For each path where water seals eliminate the potential for bypass leakage, provide a sketch to show the location of the water seal relative to the system isolation valves.

480.24 Indicate what mechanisims are available to control drywell and wetwell pressure perturbations during normal operation.

Would this system be open to the SGTS in the event of a LOCA? If so, show that the SGTS is capable of withstanding the LOCA pressure and the system filters are capable of radionuclide exposure and will still perform its intended function post-LOCA.

480.25 (6.2.3) The standby gas treatment (SGTS) is an ESF system whose effectiveness must be periodically verified as required by Appendix J to 10 CFR 50. In so doing the leakage limit of the secondary containment is measured and will be found acceptable if it agrees with the limit use in the analysis of the secondary containment depressurization time. These tests should be conducted at each refueling or at intervals not exceeding 18 months. The test limit should be consistant with the limit used for direct leakage in the analysis of the radiological consequences by the Accident Evaluation Branch (AEB). Indicate the proposed test that will be performed on the SGTS including the scheduled frequency of them, a description of the test itself, secondary containment drawdown time, the method used to measure it and the means by which the effect of open doors or hatches is included in the test program. State the design leakage rate and the SGTS fan capacity.

480.26 (6.2.4) Provide a set of piping drawings (P&ID's), which indicate the containment penetrations along with their associated piping systems. For those systems which rely on closed

loops outside primary containment as an isolation barrier indicate on the drawings the boundary of each closed loop and the grade of piping (i.e., quality group A, B, C, D) contained as part of the isolation boundary. Identify all lines connected to these closed loops which leave the secondary containment and may be bypass leak paths. Also indicate the piping connections to the closed systems and verify that these connections are valved of the under administration control so as not to break the closed loop boundary.

480.27 (6.2.4) List the penetrations which rely on a single isolation valve and a closed system outside containment as the second barrier. Verify that the following provisions of SRP 6.2.4 are satisfied for each:

1. The system reliability is greater with only one isolation valve in the line.

2. The system is closed outside containment; branch lines from the closed system must be either valved off and under administrative control or be closed systems themselves.

3. The system can accommodate a single active failure with only one isolation valve in the line.

4. The closed system outside containment must be protected from missiles, designed to seismic Category I, safety class 2 standards, and;

5. The system should have a design temperature and pressure rating at least equal to that for the containment

480.28 (6.2.4) Closed systems outside containment having a post accident function, become extensions of the containment boundary following a LOCA. Certain of these systems may also be identified as one of the redundant containment isolation barriers. Since these systems may circulate contaminated water or the containment atmosphere, system components which may leak are relied on to provide containment integrity. Therefore, discuss your plans for specifying a leakage limit for each system that becomes an extension of the containment boundary following a LOCA, and for leak testing the systems either hydrostatically or pneumatically. Discuss how the leakage will be included in the radiological assessment of the site.

480.29 (6.2.4) Describe the provisions to insure that debris will not become entrained in the purge valves and prevent their closure. Guidance is provided below which, if followed, would represent an acceptable debris screen design:

a) The debris screen should be seismic Category I and installed typically about one pipe diameter away from the inner side of the inboard isolation valve.

b) The piping between the debris screen and the valve should also be seismic Category I design.

c) The debris screen should be designed to withstand the LOCA differential pressure.

d) The debris screen openings should be about 2 inches by 1 3/16 inches.

A suggested debris screen design is enclosed as Figure 1.

480.30 (6.2.4) For those lines that are required to be open following an accident which have either two isolation valves outside containment or a single remote-manual isolation valve outside containment in a closed system show that:

1) The remote-manual isolation valve outside the containment and the piping between the containment and the valve is enclosed in a leak-tight or controlled leakage housing, or

2) the design of the piping up to and including the first remote-manual isolation valve conforms to the provisions of SRP Section 3.6.2.

In either case discuss the design of the valve and/or piping compartment in regard to the capability to detect leakage from the valve shaft and/or bonnet seals and to terminate the leakage.

480.31 (6.2.4) For each remote-manual containment isolation valve, indicate the provisions made to allow the operator in the main control room to know when to isolate by remote-manual means. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.

480.32 (6.2.4) The leak tight ECCS pump rooms are to be sized such that if a failure of a seal or gasket on a line from the suppression pool occurred inside the pump room, the volume of the suppression pool water needed to fill the room would not reduce the suppression pool level below the minimum drawdown line. Verify this by providing the details of your analysis, including the distance below the pool surface of the suction lines at the minimum drawdown level.

480.33 (6.2.4) As a result of the numerous reports on unsatsifactory performance of the resilient seats for the isolation valves in containment purge and vent lines (addressed in OIE Circular 77-11, dated September 6, 1977), Generic Issue B-20, "Containment Leakage Due to Seal Deterioration, " was established to evaluate the matter and establish an

appropriate testing frequency for the isolation valves. Excessive leakage past the resilient seats of isolation valves in purge/vent lines is typically caused by severe environmental conditions and/or wear due to frequent use. Consequently, the leakage test frequency for these valves should be keyed to the occurrence of severe environmental conditions and the use of the valves, rather than the current requirements of 10 CFR 50, Appendix J.

The staff recommends that the following provisions be added to the Technical Specifications for the leak testing of purge/vent line isolation valves:

> "Leakage integrity test shall be performed on the containment isolation valves with resilient material seals in (a) active purge/vent systems (i.e., those which may be operated during plant operating Modes 1 through 4) at least once every three months and (b) passive purge systems (i.e., those which must be administratively controlled closed during reactor operating Modes 1 through 4) at least once every six months."

In light of this, state your intention to test these butterfly valves in this manner.

480.34 (6.2.4) List the relief valves that will be used as isolation valves and verify that their setpoint is greater than 1.5 times the containment design pressure of 45 psig.

480.35 (6.2.4) SRP 6.2.4 indicates that containment isolation valves should be located as close to the containment boundary as practical (SRP pg. 6.2.4-5). FSAR Table 6.2-56 indicates that many outboard isolation valves, such as penetration Z-13, "CSH test return line to suppression pool" which has 50 ft. of pipe length from the containment, are quite a distance from the containment. List the outboard isolation valves in Table 6.2-56 which are more than 10 ft. from the containment boundary and indicate for each the reason why they must be located that distance from the containment. If no justification is available indicate why the valves should not be relocated closer to the containment.

480.36 (6.2.4) Test, vent and drain connections should be included in the Type C leak test program, i.e., these connections should be leak tested in a similar manner to the other isolation valves. Indicate your intent to comply with this provision of SRP 6.2.4.

480.37 (6.2.6) Appendix J, Section III.C.1 prescribes methods for conducting the containment isolation valve leak rate tests.

These requirements state that containment isolation valves should normally be leak tested with the test pressure applied in the same direction the valve must function to preclude leakage in an accident condition. Reverse direction testing is permitted only if it can be demonstrated that such testing yields results which are equivalent or more conservative than those obtained using same direction as post accident flow" testing. List the containment isolation valves for which Type C leak testing with reverse flow is used. For each justify by means of test data or valve design arguments that this testing is equivalent or more conservative than "same direction as post accident flow" testing.

480.38 (6.2.4) The FSAR does not specifically identify the extent of drywell-suppression chamber purging that may be necessary during normal plant operations. Discuss the manner in which Nine Mile Point 2 conforms to the requirements of Branch Technical Position CSB 6-4. Indicate how small pressure perturbations will be accommodated in the containment.

480.39 (6.2.5) Discuss the potential for a post LOCA nonuniform hydrogen concentration within the drywell and suppression chamber and the adequacy of the location of the hydrogen sampling points to obtain a representative sample. Identify the location

of the hydrogen sample points in the drywell and suppression chamber w.r.t. local structures and equipment.

480.40 (6.2.5) Indicate whether the combustible gas control system is considered an extension of the containment in post-LOCA conditions and consequently will be included within the boundary of the type A test. If so, justify the system against the criteria for closed loop systems on SRP page 6.2.3-11.

480.41 (6.2.5) Specify the manufacturer of the recombiner to be used at Nine Mile Point 2. Show that the temperature and pressure operating environment are within the recombiner qualification envelope.

480.42 (6.2.6) Valves sealed with water from a seal system may be hydrostatically tested, as opposed to air tested provided:

1. The line is not a potential containment atmosphere leak path.

2. The valve is pressurized with water to a pressure not less than 1.1 Pa.

3. The water seal must be maintained for 30 days using safety grade piping and assuming single failure of active components.

Provide a table listing valves which will be water tested due to the existence of a water leg seal. Include for each a system drawing showing the routing and elevation of the piping used to show existence of the water seal and indicate whether the provisions outlined above are satisfied. When the operation of a system is needed to maintain a water seal, e.g., the ECCS system, show that the system will keep its water seal for a sufficient period of time if the system is removed from operation. In particular indicate if jockey pumps are necessary to maintain the water leg seal and if so will the seal be maintained in the event of a failure of the pumps? Also, indicate that the liquid leakages assigned to the valves will be included in the Technical Specifications.

480.43 (6.2.6) The fluid systems listed on FSAR page 6.2-85 are listed as not being vented to the primary containment for the Type A test. Appendix J Part III A.1.d states that:

1. Systems that are required to maintain the plant in a safe condition during the test shall be operable in their normal mode and need not be vented,

2. Systems that are normally filled with water and operating under post accident conditions need not be vented.

List the systems on page 6.2-85 and indicate which criteria above is satisfied. One of the systems listed, the containment air system does not appear to meet either criteria. Justify the system and others like it which may not meet the above criteria. Indicate whether these valves listed on FSAR page 6.2-85 will be Type C tested.

480.44 (6.2.6) Provide representative drawings or sketches which depict the arrangement of the isolation valve (Type C) leakage test. These figures should clearly identify the isolation valves, and other valves which define a test volume, the location of the test taps, the location of the valves and system lines relative to the RPV, primary containment, and secondary containment.

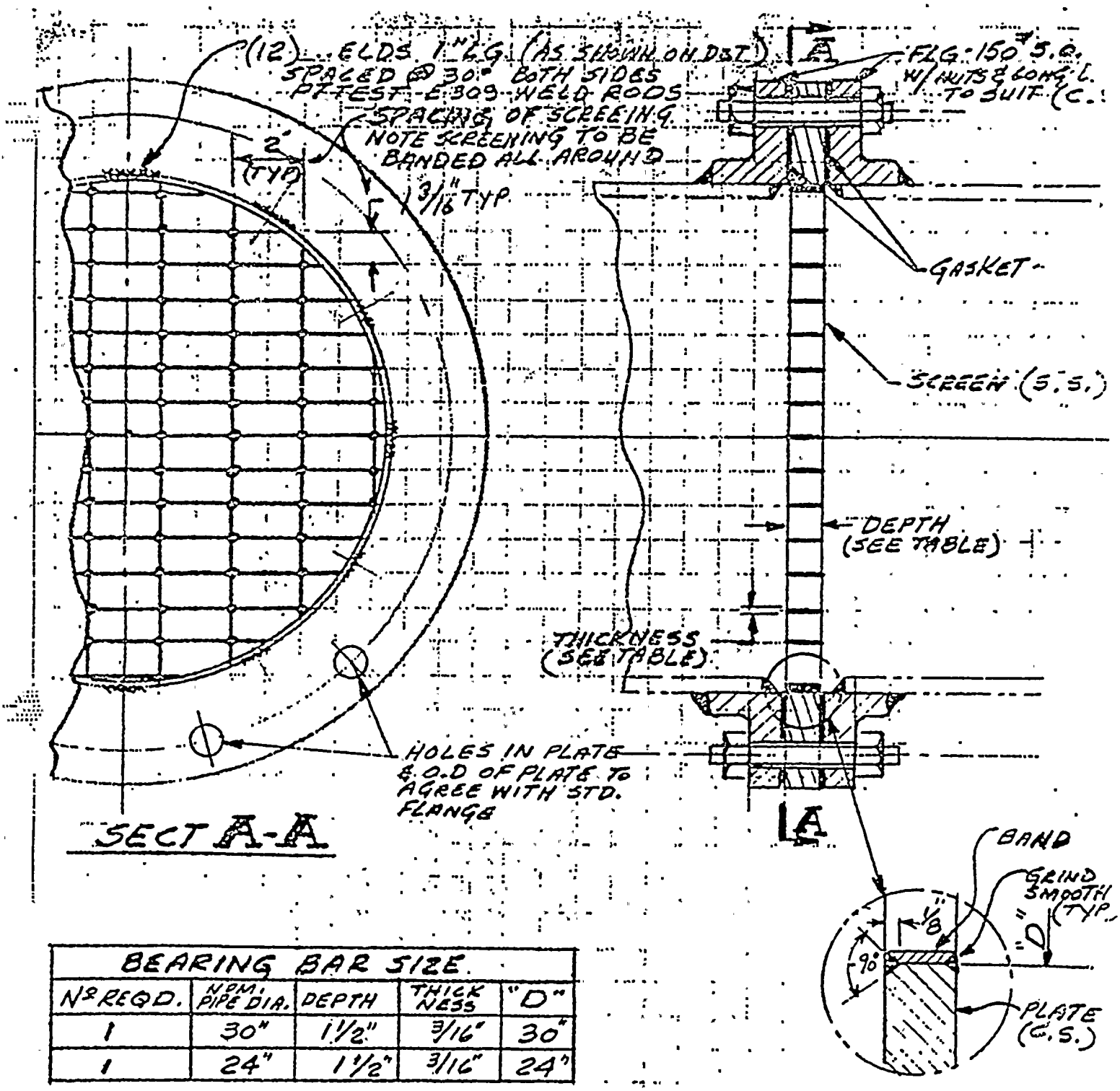
480.45 (TMI) TMI Item II.E.4.2, "Containment Isolation Dependability" has not been addressed in the Nine Mile Point 2 FSAR. Provide a schedule for the submittal of this information.

480.46 (6.2.1) Indicate the extent to which blowout panels are taken credit for in the subcompartment analyses. For each blowout panel, indicate the degree of conformance with the criteria listed below:

a. The blowout panel area and resistance as a function of time after break should be based on a dynamic analysis of the subcompartment pressure response to pipe rupture.

b. The validity of the analysis should be supported by experimental data.

c. The effect of missiles that may be generated during the transient should be considered in the safety analysis.



BEARING BAR SIZE				
Nº REQD.	NOM. PIPE DIA.	DEPTH	THICK NESS	"D"
1	30"	1 1/2"	3/16"	30"
1	24"	1 1/2"	3/16"	24"

Figure - 1

SCREEN ASS'Y & DETAIL

Attachment 13

Core Performance
REACTOR PHYSICS

QUESTION FOR NINE MILE POINT 2

491.1
SRP
15.4.7

The cited section of GESTAR II does not provide sufficient information for first cycle analyses to permit the conclusions that an acceptable evaluation of the fuel misloading event has been performed. The implication in the tabulated results is that the particular misloading event evaluated is the loading of a high enrichment bundle in a medium enrichment location. Please confirm or describe the particular event analyzed and provide arguments to show that it is the limiting event.

REQUESTS FOR ADDITIONAL INFORMATION
NINE MILE POINT NUCLEAR POWER STATION UNIT 2
INITIAL TEST PROGRAM

640.04 (14.2.7) (1.8) FSAR Subsection 14.2.7 states that conformance to Regulatory Guide 1.20 (Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing) is described in FSAR Section 1.8. FSAR Section 1.8 states that an alternative approach is described in FSAR Subsection 3.9.2.4B. Modify FSAR Subsection 3.9.2.4B or 14.2.7 to clearly describe how the alternative approach will differ from the methods specified in Regulatory Guide 1.20, and provide technical justification for the differences.

640.05 (14.2.7) Certain exceptions to Regulatory Guide 1.52 (Design, Testing, and Maintenance Criteria for Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Unit of Light-Water-Cooled Nuclear Power Plants) listed in FSAR Section 1.8 need to be deleted or modified as described below to be acceptable.

1. Modify exception to paragraph C.2.1 to delineate how the ductwork tests performed using the methods of the Associated Air Balance Council differ from the requirements given in Section 6 of ANSI N510:1975, and provide technical justification for any testing that does not address those differences.

2. Modify the exception to paragraph C.3.1 to:

   a. (See question 460.22 f).

   b. Reference or describe the normal industry practice which will be used to establish displacement criteria during balancing, and identify the criteria which you state constitutes unrealistic requirements at certain operating speeds when using the maximum vibration velocity method in ANSI N509:1976. Justify this conclusion.

   c. Provide assurance that the data contained in the AMCA certification ratings will most closely represent the manner in which the fan will be installed and operated in the appropriate system.

3. The exception to paragraph C.3.p states that Class B leakage rates shall be determined for one damper of each type instead of every damper. Delete this exception or provide expanded technical justification for not conducting leakage rate measurements for all dampers.

4. Modify FSAR Subsection 14.2.7 to reflect the level of conformance to Regulatory Guide 1.52 or to reference FSAR Section 1.8.

640.06 (14.2.7) (1.8) Delete the reference to Regulatory Guide 1.80 in FSAR Subsection 14.2.7 and substitute a statement of conformance to Regulatory Guide 1.68.3 (Preoperational Testing of Instrument and Control Air Systems) as in FSAR Section 1.8.

640.07 (1.8) Add a description of the extent of compliance with testing prescribed by NUREG-0554 "Single-Failure Proof Cranes for Nuclear Power Plants" and NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" to your statement in FSAR Section 1.8 regarding Regulatory Guide 1.104.

640.08 (14.2.7) To meet the regulatory position stated in Regulatory Guide 1.108 (Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Plants):

1. Delete your current exception to Regulatory Guide 1.108 in FSAR Subsection 14.2.7 and commit to conducting all diesel generator preoperational tests with the diesel generators installed in-plant, or provide expanded technical justification to provide assurance that vendor testing will accomplish the same test objectives as in-situ testing.

2. Delete your current exception to Regulatory Guide 1.108 (position c.2.a(3)) in FSAR Section 1.8 and commit to testing the diesel generator for two hours at a load equivalent to the 2 hour rating, not the 2000-hour rating as listed in FSAR Section 1.8.

3. Modify Preoperational Test Abstract Number 14.2-47 (Diesel Generator Mechanical System) to include testing to ensure the satisfactory operability of all check valves in the flow path of cooling water for the diesel generators from the intake to the discharge (see I&E Bulletin No. 83-03: Check Valve Failures in Raw Water Cooling Systems of Diesel Generators).

4. Modify Preoperational Test Abstract Number 14.2-97 (Emergency A-C Distribution Load Carrying Capability System) and/or Number 14.2-98 (Loss of Power/ECCS Functional Test) to demonstrate proper diesel generator operation during load shedding, including a test of the loss of the largest single load and complete loss of load, and verify that the voltage requirements are met and that the overspeed limits are not exceeded. Your testing should, in addition, provide assurance that any time delays in the diesel generator's restart circuitry will not cause the supply of compressed air used to initially rotate the engine to be consumed in the presence of a safety injection signal (see I&E Information Notice Number 83-17, March 31, 1983).

640.09 (14.2.7) Certain exceptions to Regulatory Guide 1.140 (Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants) listed in FSAR Section 1.8 need to be deleted or modified as described below to be acceptable.

1. Modify exception to paragraph C.2.f to delineate how the ductwork leak tests performed using the methods of the Associated Air Balance Council differ from the requirements given in Section 6 of ANSI N510:1975, and provide technical justification for any testing that does not address those differences.

2. Modify the exception to paragraph C.3.i to provide assurance that the data provided in the AMCA certification ratings will most closely represent the manner in which the fan will be installed in the appropriate system.

3. The exception to paragraph C.3.1 states that Class B leakage rates shall be determined for one damper of each type instead of every damper. Delete this exception or provide expanded technical justification for not conducting leakage rate measurements for all dampers.

640.10 (14.2.12) Preoperational and acceptance test abstracts use an identical format for describing what will be done and, for many of the tests, either reference an FSAR subsection that only in vague generalities discusses how the system is designed to perform, or reference an incorrect or non-existent FSAR subsection. Startup test abstracts often do not list sources of acceptance criteria. For any of the following tests subject to FSAR Chapter 17 Quality Assurance Program requirements, modify the abstract to include specific acceptance criteria or identification of the sources for the acceptance criteria to be used when test procedures are prepared. This information is necessary for the NRC inspectors who review tests procedures and evaluate test results. The test description should provide "traceability" to acceptance criteria sources such as: other FSAR subsections which contain specific detail as to the expected system performance, Technical Specifications, topical reports, vendor-furnished test specifications, and/or accident analysis assumptions.

1. Preoperational Test Abstract Numbers 14.2-45 through 51, 66, 68, 74, 76, 78, 84, 85, 92, 100, 102-105, 108, 109.

2. Startup Test Abstract Numbers 14.2-110, 112-115, 117-126, 128-137, 139-144.

640.11 (14.2.12) In accordance with the regulatory positions C.2 and C.3 of Regulatory Guide 1.41 (Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments),

1. Modify Preoperational Test Abstract Number 14.2-16 (125 V D-C Distribution) to incorporate testing to verify that at the minimum and maximum design

battery voltages, required Class 1E loads can be started and operated. The battery chargers should not be in use until after the 1E loads have started (IEEE 308-1978). For more information on problems with maximum battery voltage conditions, see I&E Information Notice 83-08, March 9, 1983.

2. Modify Preoperational Test Abstract Number 14.2-18 (115-KV Switchyard and Station Electric Feed System) and/or Number 14.2-19 (Normal A-C Distribution High Voltage System) to demonstrate the proper operation of transformer cooling under rated load or describe how data from testing under available load will be extrapolated to verify cooling capability under design loading.

3. Modify preoperational test abstracts involving sources of power to vital a-c buses to ensure that full-load testing, or extrapolation to full-load testing conditions, is accomplished.

4. Modify all preoperational test abstracts associated with d-c and on-site a-c buses to ensure that during such testing the d-c, on-site a-c, and related loads not under test will be monitored to verify absence of voltage at these buses and loads.

5. Modify any preoperational test abstract associated with d-c and on-site a-c buses where testing on Unit 2 may be dependent on Unit 1 components to ensure that independence is maintained and verified during testing.

640.12 (14.2.12) In accordance with Regulatory Guide 1.68, Appendix A.1.f, appropriate operability test should be conducted to demonstrate that waste heat rejection systems will perform as designed. Modify or reference existing preoperational test abstracts or provide additional test abstracts which will provide verification that service water pumps used for long-term post-accident core cooling will have adequate NPSH, and that there will be an absence of vortexing at the worst postulated conditions (minimum lake level and maximum lake temperature).

640.13 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.h, provide or reference preoperational test abstract descriptions in FSAR Subsection 14.2.12 that ensure that the emergency ventilation systems are capable of maintaining all Engineered Safety Features (ESF) equipment within their design temperature range with the equipment operating in a manner that will produce the maximum heat load in the compartment. If it is not practical to produce maximum heat loads in a compartment, describe the methods that will be used to develop acceptance criteria that verify design heat removal capability of emergency ventilation systems.

(Note that it is not apparent that post-accident design heat loads will be produced in ESF equipment rooms during the scheduled test phase; therefore, simply assuring that area temperatures remain within design limits during this period will not demonstrate the design heat removal capability of these systems. It will be necessary to include measurement of air and cooling water temperatures and flows, and the extrapolations used to verify that the ventilation systems can remove the postulated post-accident heat loads.)

640.14 (14.2.12) Preoperational Test Abstract Number 14.2-36 (Turbine, Reactor, and Radwaste Buildings Sampling Systems) references as acceptance criteria FSAR Subsection 11.5.1. Startup Test Abstract Number 14.2-110 (Chemical and Radiochemical) does not discuss how liquid sample holdup times will be determined. For compliance with Regulatory Guide 1.68, Appendix A.1.1, modify the preoperational test abstract, the startup test abstract, or FSAR Subsection 11.5.1 to provide a description of sample system tests that will be conducted to verify holdup times.

640.15 (14.2.12) Delete either Preoperational Test Abstract Number 14.2-39 or 14.2-69 (Reactor Building HVAC Systems) as they are currently identical.

640.16 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.h.(3), expand the Preoperational Test Abstract Number 14.2-48 (Residual Heat Removal System) test objective to include verification that the paths for the air-flow test of containment spray nozzles overlap the water-flow test paths of the pumps in order to demonstrate that there is no blockage in the flow paths.

640.17 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.h.(1)(c), modify Preoperational Test Abstract Number 14.2-48 (Residual Heat Removal System) to state that all five modes of operation will be tested.

640.18 (14.2.12) Modify Preoperational Test Abstract Number 14.2-70 (Control Building HVAC System) to include reference to testing to be conducted to meet the requirements of Regulatory Guide 1.95 (Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release). The current acceptance criteria, including the reference to FSAR Subsection 9.4.1, does not adequately establish the testing to satisfy the applicable positions of Regulatory Guide 1.95.

640.19 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.1, expand Preoperational Test Abstract Number 14.2-81 (Liquid Radwaste Handling System) and Number 14.2-104 (Solid Radwaste Handling System) to ensure that any radiation detectors and monitors which are part of those systems are tested with spiked samples of typical media, or with sources.

640.20 (14.2.12) Preoperational Test Abstract Number 14.2-87 (Plant Communications System) references FSAR Subsection 9.5.2 for acceptance criteria. Modify the preoperational test abstract or FSAR Subsection 9.5.2 to provide a description of the testing to be performed to meet the requirements of 10 CFR 50, Appendix E.IV.E, IE Bulletin No. 80-15, and Generic Letter 82-33.

640.21 (14.2.12) To comply with Regulatory Guide 1.68, Appendix A.1.c, modify Preoperational Test Abstract Number 14.2-89 (Reactor Protection System) to include or reference testing that will:

1. Account for delay times of process-to-sensor hardware (e.g., instrument lines, hydraulic snubbers).

2. Provide assurance that the response time of each primary sensor is acceptable.

3. Provide assurance that the total reactor protection system response time is consistent with your accident analysis assumptions.

NOTE: Item 2 can be accomplished by measuring the response time of each sensor during the preoperational test, stating that the response time of each sensor will be measured by the manufacturer within two years prior to fuel loading, or describing the manufacturer's certification process in sufficient detail for use to conclude that the sensor response times are in accordance with design.

640.22 (14.2.12) The acceptance criteria listed in Preoperational Test Abstract Number 14.2-90 (Shutdown from Outside the Control Room) states that the system will meet its design functions as described in FSAR Subsection 7.4. FSAR Subsection 7.4.2.4.4 states that regulatory guides that apply to the remote shutdown system are specified in Table 7.1-3. Modify Table 7.1-3 to include Regulatory Guide 1.68.2 (Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants).

640.23 (14.2.12) Startup Test Abstracts Number 14.2-110 (Chemical and Radiochemical) and 14.2-119 (APRM Calibration) do not adequately specify the test conditions at which the testing will be conducted. Modify the individual test abstracts to include the specific power level values, range of values, or test conditions at which each of the tests will be conducted, or provide a figure to indicate which tests will be conducted during each power plateau or test condition of the startup program.

640.24 (14.2.12) Modify Startup Test Abstract Number 14.2-114 (Control Rod Drive System) to clarify Level 1 and Level 2 acceptance criteria.

640.25 (14.2.10) (14.2.12) Modify FSAR Section 14.2.10 or Startup Test Abstract Number 14.2-113 to ensure that initial criticality will be approached on a startup rate of less than 1 decade/minute in accordance with Regulatory Guide 1.68 (Initial Test Program for Water-Cooled Nuclear Power Plants), Appendix A.3.

640.26 (14.2.12) Modify the acceptance criterion (Level 2, #4) in Startup Test Abstract Number 14.2-121 (RCIC System) to include verification that the FSAR-defined RCIC steam flow setpoint is consistent with the actual startup data. For more information see IE Information Notice Number 82-16: HPCI/RCIC High Steam Flow Setpoints, dated May 28, 1982.

640.27 (14.2.12) Startup Test Abstract Number 14.2-124 (System Expansion) states that certain information will be provided later. Either provide this information or state when it will be made available.

640.28 (14.2.12) Startup Test Abstract Number 14.2-133 (Full Reactor Isolation) states that full MSIV closure will be performed at TC-6 (approx. 100% power) for the first plant to start up and at 75% power for subsequent plants. Either provide technical justification for accomplishing this test at lower power levels or delete this exception to Regulatory Guide 1.68, Appendix A, 5.m.m. Note that any technical justification should include a commitment to compare test data taken with results from the first plant trip at 100% power, and should specify or reference the criteria to be used to establish that the Nine Mile Point Unit 2 results are conservative.

640.29 (14.2.12) Startup Test Abstract Number 14.2-134 (Relief Valves) states that SRV minimum capacity contains significant conservatisms in design (15-25%). We have noted on other plant startups that the capacities of the SRVs (Startup Test Abstract Number 14.2-134) and turbine bypass valves (Startup Test Abstract Number 14.2-131) are sometimes in excess of the values assumed in the accident analyses for inadvertent opening or failure of these valves. Modify the appropriate test abstracts to provide a description of the testing that demonstrates that the capacity of these valves is consistent with your accident analysis assumptions, for both the minimum and maximum capacity conditions.

640.30 (14.2.12) Provide a test description for any confirmatory in-plant tests of safety-relief valves to be performed in compliance with NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharge for BWR Plants."

640.31 (14.2.12) Startup Test Abstract Number 14.2-135 (Turbine Trip and Generator Load Rejection) states that either a turbine trip or a generator trip will be accomplished at TC-6 (approx. 100% power). In accordance with Regulatory Guide 1.68, Appendix A.5.j.j, either perform both a turbine trip and generator trip at 100% power or provide technical justification for omitting one of these tests. If one trip will not be performed, test data from previous trips and precalculated transient analyses should be extrapolated to verify adequacy of reactor dynamic response.

640.32 (14.2.12) To demonstrate the objectives of Regulatory Guide 1.68, Appendix A.5.j.j are met, modify Startup Test Abstract Number 14.2-139 (Loss of Turbine Generator and Offsite Power) to ensure that the loss of power is maintained long enough for the plant conditions to stabilize (>30 minutes).

640.33 (14.2.12) (1.10) Review the BWR Owners Group response to NUREG-0737 (as well as NUREG-0660 and NUREG-0694), Item I.G.1 (Letter from D. B. Waters to D. G. Eisenhut dated February 4, 1981) and revise the Nine Mile Point 2 FSAR Chapter 14 to include Appendix E (additional tests). Also revise FSAR Section 1.10, Item I.G.1 to include a commitment to the Owners Group recommendations.

640.34 (14.2.12) Our review of your test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68 (Revision 2) Appendix A may not be adequately demonstrated by your initial test program. Expand FSAR Subsection 14.2.12 to address the following items or explain why such preoperational or startup testing is not applicable to your facility:

NOTE: Inclusion of a test description in FSAR Chapter 14 does not necessarily imply that the test becomes subject to FSAR Chapter 17 Quality Assurance Program controls. Certain tests, performed prior to fuel loading to verify system operability, may be referred to as "acceptance tests" to distinguish them from "preoperational tests" subject to FSAR Chapter 17 test control.

ACCEPTANCE AND PREOPERATIONAL TESTS

R.G. 1.68 APPENDIX A	FSAR SECTION	DESCRIPTION
1.b(1)		Rod Block Monitors
1.d(3)	5.2.2	Relief Valves
1.d(4)	5.2.2	Safety Valves
1.e(3)	5.4.5	Main Steam Isolation Valves
1.e(6)	10.4.4	Turbine Bypass Valves
1.h	5.4.4	Main Steam Line Flow Restrictors
1.h(8)	6.3.2.2.5	ECCS Discharge Line Fill System
1.h(10)	9.2.5	Ultimate Heat Sink
1.i(10)	6.2.1.1.2	Containment and Suppression Pool Vacuum-breaker Tests
1.j(7)	7.6.1.3	ECCS Leak Detection System
1.j(12)		Failed Fuel Detection System
1.j(13)	7.2.1.2	Source Range Monitors
1.j(21)	7.7.1.1.2	Reactor Mode Switch and Associated Functions
1.j(23)	6.2.5.2.5	Hydrogen and Oxygen Analyzer System
1.l(5)	11.5.2.1.4	Condenser Offgas Isolation
1.l(7)	11.5.2.1.3	Liquid Radwaste Effluent Isolation
1.n(3)	9.4.10.2	Ventilation Chilled Water System
1.m(3)	9.1.4	Operability and leak tests of sectionalizing devices and drains and leak tests of gaskets or bellows in the refueling canal and fuel storage pool

1.m(4)	9.1.4.2	Dynamic (100%) and static (125%) load tests of cranes, hoists, and associated fuel storage and handling systems
1.m(5)	9.1.4.2	Fuel Transfer Devices
1.o(1)	9.1.4.2.2	Polar crane dynamic (100%) and static (125%) loading tests

STARTUP TESTS

2.a		Partially Loaded Core Shutdown Margin Calculation
2.c		Final Test Reactor Protection System
2.d		Final Reactor Leakrate Tests
5.g		Rod Block Monitor
5.k.		High Pressure Coolant Spray Tests
5.s	9.2.6	Hotwell Level Control System, Reactor Coolant Makeup and Letdown Systems
5.w		Containment Penetration Coolers. Provide a test description or, on those penetrations where coolers are not used, include a test description for a containment penetration concrete temperature survey to assure that penetrations will not subject concrete to temperatures over 200°F.

5.i.i	15.3	Demonstrate that the dynamic response of the plant is in accordance with design for limiting closure of reactor coolant system flow control valves. The method for initiating control valve closure should result in the fastest credible coastdown in flow.
5.g.g	15.8	ATWS Test

640.35 (14.2.5) To help facilitate approval of future changes to the Nine Mile Point 2 Initial Test Program:

1. For portions of any preoperational tests (including review and approval of test results) which are intended to be conducted after fuel loading: (a) list each test; (b) state what portions of each test will be delayed until after fuel loading; (c) provide technical justification for delaying these portions; and (d) state when each test will be completed.

2. List and provide technical justification for any tests or portions of tests described in FSAR Chapter 14 which you believe should be exempted from the license condition requiring prior NRC notification of major test changes to tests intended to verify the proper design, construction, or performance of systems, structures, or components important to safety (fulfill General Design Criteria (GDC) functions and/or are subject to 10 CFR 50 Appendix B Quality Assurance requirements).

640.13 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.h, provide or reference preoperational test abstract descriptions in FSAR Subsection 14.2.12 that ensure that the emergency ventilation systems are capable of maintaining all Engineered Safety Features (ESF) equipment within their design temperature range with the equipment operating in a manner that will produce the maximum heat load in the compartment. If it is not practical to produce maximum heat loads in a compartment, describe the methods that will be used to develop acceptance criteria that verify design heat removal capability of emergency ventilation systems.

(Note that it is not apparent that post-accident design heat loads will be produced in ESF equipment rooms during the scheduled test phase; therefore, simply assuring that area temperatures remain within design limits during this period will not demonstrate the design heat removal capability of these systems. It will be necessary to include measurement of air and cooling water temperatures and flows, and the extrapolations used to verify that the ventilation systems can remove the postulated post-accident heat loads.)

640.14 (14.2.12) Preoperational Test Abstract Number 14.2-36 (Turbine, Reactor, and Radwaste Buildings Sampling Systems) references as acceptance criteria FSAR Subsection 11.5.1. Startup Test Abstract Number 14.2-110 (Chemical and Radiochemical) does not discuss how liquid sample holdup times will be determined. For compliance with Regulatory Guide 1.68, Appendix A.1.1, modify the preoperational test abstract, the startup test abstract, or FSAR Subsection 11.5.1 to provide a description of sample system tests that will be conducted to verify holdup times.

640.15 (14.2.12) Delete either Preoperational Test Abstract Number 14.2-39 or 14.2-69 (Reactor Building HVAC Systems) as they are currently identical.

640.16 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.h.(3), expand the Preoperational Test Abstract Number 14.2-48 (Residual Heat Removal System) test objective to include verification that the paths for the air-flow test of containment spray nozzles overlap the water-flow test paths of the pumps in order to demonstrate that there is no blockage in the flow paths.

640.17 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.h.(1)(c), modify Preoperational Test Abstract Number 14.2-48 (Residual Heat Removal System) to state that all five modes of operation will be tested.

640.18 (14.2.12) Modify Preoperational Test Abstract Number 14.2-70 (Control Building HVAC System) to include reference to testing to be conducted to meet the requirements of Regulatory Guide 1.95 (Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release). The current acceptance criteria, including the reference to FSAR Subsection 9.4.1, does not adequately establish the testing to satisfy the applicable positions of Regulatory Guide 1.95.

640.19 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.1.1, expand Preoperational Test Abstract Number 14.2-81 (Liquid Radwaste Handling System) and Number 14.2-104 (Solid Radwaste Handling System) to ensure that any radiation detectors and monitors which are part of those systems are tested with spiked samples of typical media, or with sources.

640.20 (14.2.12) Preoperational Test Abstract Number 14.2-87 (Plant Communications System) references FSAR Subsection 9.5.2 for acceptance criteria. Modify the preoperational test abstract or FSAR Subsection 9.5.2 to provide a description of the testing to be performed to meet the requirements of 10 CFR 50, Appendix E.IV.E, IE Bulletin No. 80-15, and Generic Letter 82-33.

640.21 (14.2.12) To comply with Regulatory Guide 1.68, Appendix A.1.c, modify Preoperational Test Abstract Number 14.2-89 (Reactor Protection System) to include or reference testing that will:

1. Account for delay times of process-to-sensor hardware (e.g., instrument lines, hydraulic snubbers).

2. Provide assurance that the response time of each primary sensor is acceptable.

3. Provide assurance that the total reactor protection system response time is consistent with your accident analysis assumptions.

NOTE: Item 2 can be accomplished by measuring the response time of each sensor during the preoperational test, stating that the response time of each sensor will be measured by the manufacturer within two years prior to fuel loading, or describing the manufacturer's certification process in sufficient detail for use to conclude that the sensor response times are in accordance with design.

640.22 (14.2.12) The acceptance criteria listed in Preoperational Test Abstract Number 14.2-90 (Shutdown from Outside the Control Room) states that the system will meet its design functions as described in FSAR Subsection 7.4. FSAR Subsection 7.4.2.4.4 states that regulatory guides that apply to the remote shutdown system are specified in Table 7.1-3. Modify Table 7.1-3 to include Regulatory Guide 1.68.2 (Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants).

640.23 (14.2.12) Startup Test Abstracts Number 14.2-110 (Chemical and Radiochemical) and 14.2-119 (APRM Calibration) do not adequately specify the test conditions at which the testing will be conducted. Modify the individual test abstracts to include the specific power level values, range of values, or test conditions at which each of the tests will be conducted, or provide a figure to indicate which tests will be conducted during each power plateau or test condition of the startup program.

640.24 (14.2.12) Modify Startup Test Abstract Number 14.2-114 (Control Rod Drive System) to clarify Level 1 and Level 2 acceptance criteria.

640.25 (14.2.10) (14.2.12) Modify FSAR Section 14.2.10 or Startup Test Abstract Number 14.2-113 to ensure that initial criticality will be approached on a startup rate of less than 1 decade/minute in accordance with Regulatory Guide 1.68 (Initial Test Program for Water-Cooled Nuclear Power Plants), Appendix A.3.

640.26 (14.2.12) Modify the acceptance criterion (Level 2, #4) in Startup Test Abstract Number 14.2-121 (RCIC System) to include verification that the FSAR-defined RCIC steam flow setpoint is consistent with the actual startup data. For more information see IE Information Notice Number 82-16: HPCI/RCIC High Steam Flow Setpoints, dated May 28, 1982.

640.27 (14.2.12) Startup Test Abstract Number 14.2-124 (System Expansion) states that certain information will be provided later. Either provide this information or state when it will be made available.

640.28 (14.2.12) Startup Test Abstract Number 14.2-133 (Full Reactor Isolation) states that full MSIV closure will be performed at TC-6 (approx. 100% power) for the first plant to start up and at 75% power for subsequent plants. Either provide technical justification for accomplishing this test at lower power levels or delete this exception to Regulatory Guide 1.68, Appendix A, 5.m.m. Note that any technical justification should include a commitment to compare test data taken with results from the first plant trip at 100% power, and should specify or reference the criteria to be used to establish that the Nine Mile Point Unit 2 results are conservative.

640.29 (14.2.12) Startup Test Abstract Number 14.2-134 (Relief Valves) states that SRV minimum capacity contains significant conservatisms in design (15-25%). We have noted on other plant startups that the capacities of the SRVs (Startup Test Abstract Number 14.2-134) and turbine bypass valves (Startup Test Abstract Number 14.2-131) are sometimes in excess of the values assumed in the accident analyses for inadvertent opening or failure of these valves. Modify the appropriate test abstracts to provide a description of the testing that demonstrates that the capacity of these valves is consistent with your accident analysis assumptions, for both the minimum and maximum capacity conditions.

640.30 (14.2.12) Provide a test description for any confirmatory in-plant tests of safety-relief valves to be performed in compliance with NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharge for BWR Plants."

640.31 (14.2.12) Startup Test Abstract Number 14.2-135 (Turbine Trip and Generator Load Rejection) states that either a turbine trip or a generator trip will be accomplished at TC-6 (approx. 100% power). In accordance with Regulatory Guide 1.68, Appendix A.5.j.j, either perform both a turbine trip and generator trip at 100% power or provide technical justification for omitting one of these tests. If one trip will not be performed, test data from previous trips and precalculated transient analyses should be extrapolated to verify adequacy of reactor dynamic response.

640.32 (14.2.12) To demonstrate the objectives of Regulatory Guide 1.68, Appendix A.5.j.j are met, modify Startup Test Abstract Number 14.2-139 (Loss of Turbine Generator and Offsite Power) to ensure that the loss of power is maintained long enough for the plant conditions to stabilize (>30 minutes).

640.33 (14.2.12) (1.10) Review the BWR Owners Group response to NUREG-0737 (as well as NUREG-0660 and NUREG-0694), Item I.G.1 (Letter from D. B. Waters to D. G. Eisenhut dated February 4, 1981) and revise the Nine Mile Point 2 FSAR Chapter 14 to include Appendix E (additional tests). Also revise FSAR Section 1.10, Item I.G.1 to include a commitment to the Owners Group recommendations.

640.34 (14.2.12) Our review of your test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68 (Revision 2) Appendix A may not be adequately demonstrated by your initial test program. Expand FSAR Subsection 14.2.12 to address the following items or explain why such preoperational or startup testing is not applicable to your facility:

NOTE: Inclusion of a test description in FSAR Chapter 14 does not necessarily imply that the test becomes subject to FSAR Chapter 17 Quality Assurance Program controls. Certain tests, performed prior to fuel loading to verify system operability, may be referred to as "acceptance tests" to distinguish them from "preoperational tests" subject to FSAR Chapter 17 test control.

ACCEPTANCE AND PREOPERATIONAL TESTS

R.G. 1.68 APPENDIX A	FSAR SECTION	DESCRIPTION
1.b(1)		Rod Block Monitors
1.d(3)	5.2.2	Relief Valves
1.d(4)	5.2.2	Safety Valves
1.e(3)	5.4.5	Main Steam Isolation Valves
1.e(6)	10.4.4	Turbine Bypass Valves
1.h	5.4.4	Main Steam Line Flow Restrictors
1.h(8)	6.3.2.2.5	ECCS Discharge Line Fill System
1.h(10)	9.2.5	Ultimate Heat Sink
1.i(10)	6.2.1.1.2	Containment and Suppression Pool Vacuum-breaker Tests
1.j(7)	7.6.1.3	ECCS Leak Detection System
1.j(12)		Failed Fuel Detection System
1.j(13)	7.2.1.2	Source Range Monitors
1.j(21)	7.7.1.1.2	Reactor Mode Switch and Associated Functions
1.j(23)	6.2.5.2.5	Hydrogen and Oxygen Analyzer System
1.l(5)	11.5.2.1.4	Condenser Offgas Isolation
1.l(7)	11.5.2.1.3	Liquid Radwaste Effluent Isolation
1.n(3)	9.4.10.2	Ventilation Chilled Water System
1.m(3)	9.1.4	Operability and leak tests of sectionalizing devices and drains and leak tests of gaskets or bellows in the refueling canal and fuel storage pool

1.m(4)	9.1.4.2	Dynamic (100%) and static (125%) load tests of cranes, hoists, and associated fuel storage and handling systems
1.m(5)	9.1.4.2	Fuel Transfer Devices
1.o(1)	9.1.4.2.2	Polar crane dynamic (100%) and static (125%) loading tests

STARTUP TESTS

2.a		Partially Loaded Core Shutdown Margin Calculation
2.c		Final Test Reactor Protection System
2.d		Final Reactor Leakrate Tests
5.g		Rod Block Monitor
5.k		High Pressure Coolant Spray Tests
5.s	9.2.6	Hotwell Level Control System, Reactor Coolant Makeup and Letdown Systems
5.w		Containment Penetration Coolers. Provide a test description or, on those penetrations where coolers are not used, include a test description for a containment penetration concrete temperature survey to assure that penetrations will not subject concrete to temperatures over 200°F.

5.i.i	15.3	Demonstrate that the dynamic response of the plant is in accordance with design for limiting closure of reactor coolant system flow control valves. The method for initiating control valve closure should result in the fastest credible coastdown in flow.
5.g.g	15.8	ATWS Test

640.35 (14.2.5) To help facilitate approval of future changes to the Nine Mile Point 2 Initial Test Program:

1. For portions of any preoperational tests (including review and approval of test results) which are intended to be conducted after fuel loading: (a) list each test; (b) state what portions of each test will be delayed until after fuel loading; (c) provide technical justification for delaying these portions; and (d) state when each test will be completed.

2. List and provide technical justification for any tests or portions of tests described in FSAR Chapter 14 which you believe should be exempted from the license condition requiring prior NRC notification of major test changes to tests intended to verify the proper design, construction, or performance of systems, structures, or components important to safety (fulfill General Design Criteria (GDC) functions and/or are subject to 10 CFR 50 Appendix B Quality Assurance requirements).

NINE MILE POINT UNIT 2

GENERIC ISSUES

REQUEST FOR INFORMATION

730.1 The Atomic Safety and Licensing Board in ALAB-444 determined that the Safety Evaluation Report for each plant should contain an assessment of each significant unresolved generic safety question. It is the staff's view that the generic issues identified as "Unresolved Safety Issues" (NUREG-0606) are the substantive safety issues referred to by the Appeal Board. Accordingly, we are requesting that you provide your justification for permitting plant operation pending resolution of these issues. This should include a description of any interim measures in terms of design or operating procedures or investigative programs that are being pursued to address these concerns. The justification should provide an overall summary of your position on each issue rather than a reference to various sections of the SER where related information is presented.

There are currently a total of 27 Unresolved Safety Issues. We do not require information from you at this time for a number of the issues since a number of the issues do not apply to your type of reactor or because a generic resolution has been issued. Issues which have been resolved have been or are being incorporated into the NRC licensing guidance and are addressed as a part of the normal review process. However, we do request the information noted above for each of the issues listed below:

1. Water Hammer (A-1)
2. Systems Interaction (A-17)
3. Seismic Design Criteria (A-40)
4. Containment Emergency Pump Performance (A-43)
5. Station Blackout (A-44)
6. Shutdown Decay Heat Removal Requirements (A-45)
7. Safety Implications of Control Systems (A-47)
8. Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)