Docket No. 50-410

NOV 3 0 1972

Niagara Mohawk Power Corporation
ATTN: Mr. Thomas J. Brosnan
Vice President & Chief Engineer
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

In order that we may continue our review of your application for a license to construct the Nine Mile Point Nuclear Station Unit 2 (NMPNS-2) additional information is required. The information requested is described in the enclosure and pertains to the different sections of the Preliminary Safety Analysis Report (PSAR). The sections are the following: 2.0 Site, 3.0 Reactor, 4.0 Reactor Coolant System, 6.0 Core Standby Cooling Systems, 7.0 Control and Instrumentation, 8.0 Electrical Systems, 12.0 Station Structures and Shielding, 14.0 Unit 2 Safety Analysis, and Appendix C Equipment Design Criteria.

The questions in the enclosure have been grouped by sections that correspond to the relevant sections of the PSAR and are numbered in seriatim with previous requests for information.

In order to maintain our licensing review schedule, we will need a completely adequate response to all enclosed questions by January 29, 1973. Please inform us within 7 days after receipt of this letter of your confirmation of the schedule date or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our request, it is highly likely that the overall schedule for completing the licensing review for the project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning tbithis project, the extent of the extension will most likely be greater than the delay in your response.

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Please contact us if you have any questions regarding the information requested.

Sincerely,

Original signed by
Steven A. Varga for

Robert A. Clark, Chief
Gas Cooled Reactors Branch
Directorate of Licensing

Enclosure:
Request for Additional
Information

cc: LeBoeuf, Lamb, Leiby
& MacRae
ATTN: Arvin E. Upton, Esq.
1821 Jefferson Place, N. W.
Washington, D. C. 20036

DISTRIBUTION:
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GCR Reading
S. Hanauer
R. S. Boyd
D. Skovholt
F. Schroeder
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D. Knuth
R. Tedesco
H. Denton
BWR Branch Chiefs
W. Haass
OGC
RO (3)
A. Bournia
H. Gearin
V. Stello
R. Frahm
R. Kirkwood
W. Gammill
R. McMullen
J. Kohler
R. Kasnasiewicy
E. Markee
L. Hulman
E. Hawkins
L. Shao
R. Lipinski
V. Moore
M. Dunenfeld
F. AShe
F. Rosa

OFFICE ►	L:GCR	L:GCR				
SURNAME ►	ABournia;jkm	RClark				
DATE ►	11/29/72	11/ /72				

Docket

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-410

NOV 30 1972

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REQUEST FOR ADDITIONAL INFORMATION

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION-2

DOCKET NO. 50-410

2.0 SITE

2.1 Discuss your plans to control the exclusion area where it encircles a portion of Lake Ontario during an emergency.

2.2 Locate all commercial air corridors with respect to the plant.

2.3 Describe in detail any hazardous products in the site vicinity in transit by either rail, water or highway.

2.4 Describe the peak traffic density that can be anticipated on Highway 29 and Lakeview Road.

2.5 A Design Bases Earthquake (now called Safe Shutdown Earthquake [SSE]) acceleration of .15g was utilized for the Fitzpatrick Power Plant to provide a sufficient level of conservatism to cover many unknowns regarding the regional structural geology. Part of the justification in support of the utilization of a lower SSE acceleration at Nine Mile Point Unit 2 is presented in reference 1, Dames and Moore "Regional Geologic and Tectonic Study, St. Lawrence River Valley," for New York State Atomic and Space Development Authority, 1971. Our review of this document and supporting data has not been completed. During a meeting held on 9 November, 1971 among New York State ASDA, Dames and Moore, AEC, NOAA, and USGS, additional studies and documentation were requested to support conclusions presented in the report. These data have not been provided. Because of our continued concern regarding the possible extension of the St. Lawrence Rift Structure to the Southwest beyond the Ottawa-Bonnechere structure, and to justify using an SSE value lower than that recommended for Fitzpatrick, it is necessary that the data requested on 9 November, 1971 be provided. These additional items requested which are pertinent to the Nine Mile Point site areas are as follows:

a. A geologic map drawn at a convenient scale that ties together the known on-shore geology, both regional and local (based on data available such as geologic maps, borings, aeromagnetic data, etc.), with the seismic reflection profiles and the field mapping accomplished during this investigation is needed.

b. Expand the present profile interpretations and discussions to include a documentation of the amounts of stratigraphic displacements. This is needed because the offsets shown in the report only represent discontinuities between certain reflecting horizons. The reflecting mediums should be correlated with specific stratigraphic horizons.

c. More documentation should be presented regarding the subsurface conditions on both sides of the river. Such documentation could be summary geologic profiles, based on boring logs and field mapping. Geologic cross sections of the St. Lawrence River should be included.

d. Dames and Moore postulates the extension of the Gloucester Fault to the St. Lawrence River. If it can be shown that a portion of the St. Lawrence Rift does not continue down the St. Lawrence River past the site, then the length of this Fault is now significant in demonstrating the conservatism of the proposed SSE. Discuss the Gloucester Fault in detail and provide substantiation that it extends only to the north side of the Ottawa Graben and does not form a boundary of the Bonnechere and Ottowa Grabens and thus continue for several hundred miles. Also, include in this submittal a discussion of the Kemptville Fault and its regional relationships and include appropriate documentation.

e. The Dames and Moore investigations and report have attempted to show that there is no evidence of the St. Lawrence Rift System in the upper Paleozoic bedrock strata beyond the south boundary of the Ottawa Graben. With the submission of the additional data requested we expect that this will be accomplished. However, the question of whether or not a deep-seated linear zone of deformation exists, which is reflected at the surface

by the linearity of the St. Lawrence River, must be addressed. This discussion should be documented with whatever information is available, such as deep well logs, gravity, or aeromagnetic data, etc. This discussion should also provide a plausible explanation for the linearity of the St. Lawrence River southwest of Montreal.

2.6 Several minor tectonic deformation features have been identified in the site vicinity during various investigations. It is acknowledged in the PSAR that similar features may be present beneath the Nine Mile Point Unit 2 site. In view of this possibility, if a construction permit is granted, we will require geological maps and logs of all excavation for Category I structures prior to actual construction.

2.7 Identify and evaluate any unrelieved residual stresses in the foundation bedrock.

2.8 Describe the ability of all safety-related facilities to withstand severe local rainfall up to and including probable maximum precipitation, as defined by "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6 - 48 Hours," Hydrometeoroligical Report No. 33; U. S. Weather Bureau (now NOAA); 1956. Among the facilities that should be considered for this condition are (a) the local and interior drainage system and its effectiveness to protect the safety-related facilities; and (b) roofs and drainage of safety-related buildings.

2.9 Determine the probable maximum surge (still water level) and related wind-generated wave action by using probable maximum wind velocities of 100 miles per hour (mph) in lieu of the 88 mph wind used in the PSAR. The 100 mph maximum wind, associated meteorological parameters, and wind field configuration and orientation are considered the probable maximum winds for over-water wind conditions for the Great Lakes area. This maximum 100 mph wind velocity with the probable maximum wind fields in the approved analysis for James A. Fitzpatrick Nuclear Power Plant (located approximately 0.5 mile east of the subject site) resulted in a still water level of 1.5 foot higher (elev. 254.0, USLS 1935) than that computed when a maximum wind velocity of 88 mph was used.

2.10 Discuss the ability of safety-related equipment to function in the event of the probable minimum low water set down. Provide an estimate of the head loss through the intake conduit and the intake sump for such a condition, and compare the resulting water sump level with the required minimum levels for the service water pumps. If the required level is less than the minimum sump level, discuss plans to prevent a loss of cooling water.

2.11 Clarify the discussion of wave characteristics (par. 2.7.8.3.1). Furnish details as to the derivation and computation of the wave producing the maximum force used in the analysis (non-breaking, breaking or broken) on the intake structure. Provide the analysis of the dike based on Section 4.27 of Technical Report No. 4 "Shore Protection, Planning and Design,"1966, considering the wave that will just break on the dike structure. Preliminary independent calculations (by our consultant the U. S. Army Coastal Engineering Research Center - CERC) of maximum wave runup on the structure and overtopping of the dike based on a stillwater elevation of 254.0 USLS 1935 indicate that there would be about 20 feet of runup and in order to prevent over topping, the dike should be essentially impermeable to about elevation 274.0 USLS 1935. If the crest of the dike is maintained at elevation 263.0, USLS 1935, over topping will occur and the average flow over the dike due to this condition would be about 3 cubic feet per second (cfs) per linear foot.

2.12 Does safe plant operation, or maintainence of a shutdown mode, depend on the integrity of the dike structure between the lake front and the plant? In the event the dike is required to protect safety-related facilities, consideration should be given to severe surge and associated wind-generated wave action (runup) up to and including probable maximum conditions.

a. Describe the integrity of the dike with respect to the hydrodynamic stability and the basis of design for the riprap, including type and weight of the armor units.

b. Potential overtopping should be evaluated based on the most severe breaking wave resulting from critical combinations of wave height and period producing the maximum runup on the dike structure, and not necessarily from the runup resulting from the maximum wave. If overtopping or splash-over occurs, evaluate its effects and potential flooding on safety-related facilities.

2.13 Supplement 11 of the FSAR for James A. Fitzpatrick Nuclear Power Plant gave the technical bases for both the one-dimensional and two-dimensional surge models used to predict the maximum and minimal probable set up. These technical bases should be included as part of this PSAR, or properly referenced to the James A. Fitzpatrick Nuclear Power Plant FSAR.

2.14 Provide the location, type of use, owner, demand, depth, top casing elevation and water elevation of all wells within about two miles of the site, or appropriately reference the information furnished in Supplement 5 of the FSAR for James A. Fitzpatrick Nuclear Power Plant, if it is applicable.

2.15 Provide joint frequency distributions of wind speed and direction by ΔT stability class for a representative one (preferable two) full year period from onsite data. Specify the ΔT and wind speed classes as shown in Tables 1 and 2 of Safety Guide 23. One set of tables should be based on winds measured at the 31-ft elevation; a second set of tables should be based on winds measured at the 203-ft level. Lapse rates should be based on your measurements made between 30 ft and 202 ft. Specify the percentage of data recovery for the period of record. Where periods of missing data are of days duration (as opposed to sporadic duration of a few hours at a time), specify the periods of missing data. Present any evidence as to the degree of representativeness of the period of data collection.

2.16 Safety Guide 23 and portions of 10 CFR referenced therein point out the need for onsite meteorological measurements in the assessment of consequences of accidental or routine emissions to the atmosphere during the operating phase of the power reactor. Describe and discuss your plans for a continued onsite meteorological program.

2.17 Discuss the elevation of the stack top with regard to the elevation of the topography for distances out to at least five miles. Assuming the effective stack height to be the difference between stack top elevation and topography elevation, provide a table of effective stack height versus distance for each of the sixteen 22.5 degree direction segments. Identify, in each increment of distance, the elevation of the highest point; its height with respect to the top of the stack; and its distance from the stack in meters. Considering the effective stack heights thus obtained, in conjunction with the distances and meteorological data collected at the site, provide the annual average χ/Q values as a function of distance from the stack out to five miles. It would

be desirable to present the data in a format similar to Table 2.2.1 and Figure 2.2.13 of the Peach Bottom FSAR (Docket No. 50-278).

2.18 Provide a detailed, scaled map of all effluent release points. If an effluent release point is elevated, provide the elevation above ground level. Provide the stack height, stack exit diameter and stack gas velocity for all effluent stacks.

3.0 REACTOR

3.1 In the last paragraph on page 3.3-3 it states that the actual replacement of control rods depends on the loss of reactivity control capability and helium pressure buildup. These vary with control rod position. How will these two facets be measured for each control rod and what are the criteria to be used for actual control rod replacement.

3.2 The information on thermal-hydraulic information does not adequately present what axial and radial power peaking factors and axial power distribution were used in the LOCA analysis. Please provide the necessary information.

3.3 In Section 3.5.6.1, a discussion is given on determining the pressure drop in the core. Please provide the following:

(a) Correlation used for the two phase frictional multiplier.

(b) Experimental information available to confirm that the two phase multiplier is identical for determining the pressure drop as for the friction pressure drop.

(c) References or analyses that establish the total acceleration pressure drop is only a few percent of the total pressure drop.

3.4 The PSAR should present all aspects of the nuclear design and operation of the reactor core in sufficient detail to permit a reviewer to follow, understand and perform check-type calculations of details, as well as gross features so that potential errors or misunderstandings can be spotted and margins of safety estimated. This necessitates the inclusion of a complete, detailed, and up-to-date presentation of all design and operating aspects of the core. Such a presentation must include models and their justification (a list of code names would be insufficient), information covering the full range of operating conditions of power, time in cycle, and cycle, and information on the error range of all discussed parameters and the experimental justification of this error range. Based on our review, the nuclear design information provided in the PSAR does not satisfy these requirements.

The following provides a partial listing of examples of additional information needed in the area of nuclear design:

a. The scram function curves shown in Figure 3.4-10 and 3.4-11 are known to be out-of-date and incorrect. These should be revised and information provided as to how and under what reactor conditions scram curves are calculated, their axial shape and magnitude, dependence on power, flow, time in cycle, and cycle, the uncertainty assigned to these calculations and justification for this uncertainty, and the relationship of this information to the function(s) used in the transient analyses.

b. Update your xenon transient information with suitable references to modern topical reports and experiments.

c. Provide additional information relative to the moderator density and temperature coefficient discussions including more precise calculational techniques rather than approximate formulae, the quantitative and qualitative differences between inter- and intra- assembly moderator coefficients, the error or uncertainty range and the experimental verification accomplished on operating reactors (note the Monticello reactivity anomaly situation), non-uniform density change effects in transients, and the relationship of this information to that used in transient analyses.

d. Provide additional detailed information on the gadolinia fuel pins relative to shutdown margins, power densities throughout life, and lowered damage limits. Also provide a safety evaluation relating reactivity and thermal margins to design criteria, the effects of absorber omission, and fuel loading errors. Comparison data should be provided relative to the design of gadolinia fuel pins used in other operational proposed reactors, and experimental data obtained from other operational facilities. Demonstrate by comparison with UO_2 rods, that the thermal limits proposed in the PSAR are controlled by either the UO_2 or the gadolinia bearing fuel rods.

e. Full information should be provided concerning power distributions in the core including both local and gross radial and axial power densities for many representative and extreme conditions of power, rod patterns (including anomalies), time in cycle, cycle, etc., and with details of error ranges, and experimental confirmations including results from recently operating reactors and checks to be made in the future.

f. Provide complete information on rod patterns including both startup and at power configurations, deviations to be allowed, and specific identifications of high reactivity worth anomalies.

g. Provide gross moderator density distributions, gross burnup patterns, and various reactivity inventories (e.g., void defects).

4.0 REACTOR COOLANT SYSTEM

4.12 In Section 4.2.4, it is stated that the recirculation pump and valves are designed considering the transient specified in Table 4.1-2. However, as indicated in Section H.2.54, a pump overspeed can occur during blowdown and to prevent this from occurring a decoupling device can be used. Provide information on this device along with testing results. In addition, supply information and calculations on the pump overspeed conditions.

4.13 For the shutdown cooling and reactor vessel head spray system described in Section 4.7.5.2, submit the pressures associated with the temperature-time quantities stated.

4.14 What provisions are incorporated in the design of the HPCS, LPCS, and LPCI systems to assure that a break in a pump suction line between the suppression chamber and the first external isolation valve will not result in the loss of ECCS functions.

4.15 For leakage detection some systems are temperature sensed in the same general area. For example, in the equipment ventilation area where three of these systems are located, can one distinguish which detection system among RCIC, RHR or RWC systems has set the alarm off when a high temperature differential alarm is sounded?

4.16 Describe the method of analyses used and show the variation of liquid water rate delivered by the RCIC system during shutdown. In conjunction show the steam supply rate to the RCIC steam-driven turbine-pump.

4.17 Show that NPSH for the Core Standby Cooling System pumps are met during operational and LOCA transients.

6.0 CORE STANDBY COOLING SYSTEMS

6.1 Define "one fuel time constant" (page 6.5-13) and explain how it is calculated.

6.2 Provide a break spectrum analysis for the steam line break similar to the liquid line break spectrum shown on Figure 6.5-23.

6.3 Clarify the differences between Tables 6.3-1 and 1.5-1 (page 3 of 4) on flow rates for LPCS, HPCS and LPCI.

6.4 Discuss the capability for isolating low pressure systems from the reactor coolant system (such as the low pressure residual heat removal system from the reactor coolant system) and the diversity in equipment that closes or prevents the opening of the isolation valves on high reactor pressure.

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Provide the information requested in the Commission's Information Guide 2, "Instrumentation and Electrical Systems" (10/27/71). If the requested information is presently contained in the PSAR, or added by amendment, the response should identify the specific location of the information.

a. In your response to Question 1.c of Information Guide 2, describe the degree of conformance of your protection system to the provisions of IEEE Std 338-1971, "IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems." Describe and justify any exceptions to this standard.

b. In your response to Question 3 of Information Guide 2, describe the degree of conformance of your seismic testing program to IEEE Std 344-1971, "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Station." Describe and justify any exceptions to this standard.

c. In your response to Question 4 of Information Guide 2, describe the degree of conformance to IEEE Std 336-1971, "IEEE Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations." Describe and justify any exceptions to this standard.

d. In your response to Question 7 of Information Guide 2, include all safety related equipment and components located in the reactor building (the secondary containment) which must operate during or subsequent to a steamline break or feedwater line break accident. State the limiting conditions of operation for these equipments.

e. In your response to Question 7 of Information Guide 2, describe the degree of conformance to IEEE Standard 317-1971 "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations." Identify and justify any exceptions to this standard.

f. In your response to Question 8 of Information Guide 2, describe the degree of conformance of your safety related instrumentation, control and electrical equipment to the provisions of IEEE 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations." Identify and justify any exceptions to this standard.

g. In your response to Question 12 of Information Guide 2, refer to IEEE Std 308-1971. (Note: Where a conflict exists between the "eight-hour" provision of Section 5.2.3.4 of IEEE Std 308-1971 and General Design Criterion 17, the applicable provisions of Criterion 17 govern).

7.2 Section 50.55a of 10 CFR 50 requires that the protection system meet the requirements of the revision of IEEE 279 in effect 12 months prior to the date of issuance of a construction permit. The revision applicable to Nine Mile Point Unit 2 is IEEE Std 279-1971. Therefore, you are requested to:

a. Revise PSAR Section 7 to reflect the applicability of IEEE Std 279-1971. It should be noted that conformance with the requirements (not merely the intent) of IEEE Std 279-1971 is required. Your response should cover all protection subsystems, i.e., those supplied by General Electric as well as those supplied by other vendors.

b. Compare your design of the circuitry used to manually initiate reactor trip with that required by Section 4.17 of IEEE Std 279-1971. Based on our review, we have concluded that although means are provided to manually initiate reactor trip at the system level, the design is susceptible to single failures that would prevent manual initiation of reactor trip.

c. Compare your design with that required by Section 4.17 of IEEE Std 279-1971 regarding the means provided to manually initiate the operation of engineered safety features. It appears that the only means provided to manually initiate protective actions (emergency core cooling, containment isolation, etc.) are the manual control switches for each component (pump, valve, etc.). We do not consider this arrangement to be the "means for manual initiation of each protective action at the system level" required by IEEE Std 279-1971.

7.3 The first paragraph on Page 7.1-5 of the PSAR appears to state that both backup scram pilot valves must be energized before the air supply to the scram valves is vented thus initiating the insertion of all rods by means of this backup scheme. However, the valve configuration (backup scram pilot valves C12-F110A and C12-F110B, and the check valve in parallel with C12-F110A) shown in Figure 7.1-1 indicates that only one backup scram pilot valve need be energized to vent the scram valves. Resolve this inconsistency. Describe the operation of the system including the function of the check valve.

7.4 Describe in more detail the design features provided to insure proper operation of RHR valves MO-F008 and MO-F009 (Figure 7.3.10a). An acceptable design would include the following features:

(a) At least two valves in series, with each valve interlocked to prevent valve opening unless the primary system pressure is below the RHR system design pressure.

(b) Interlocks of diverse principles, and designed to meet the requirements of IEEE 279.

(c) Automatic closure of the valves whenever the primary coolant system pressure exceeds a selected fraction of the RHR system design pressure. These closure devices should be designed to meet the requirements of IEEE 279.

If the design, or planned modifications, are based on criteria different from those above, discuss your reasons for concluding that the design provides an equivalent degree of protection.

7.5 The control schemes for the LPCS and LPCI systems include an interlock which prevents automatic opening of the injection valves until reactor pressure drops to a preselected value, as sensed by a differential pressure switch connected across the valves. Discuss the purpose of this feature. If the interlock is not required for safety, the disucssion should address the effects of its inclusion on the reliability of the LPCS and LPCI systems. Your response should also include a decription of the testability provided in the design and your intent with regard to periodic testing of this interlock feature.

7.6 In Section 7.5 it is stated that single-failure-proof circuitry is provided to sense "all rods inserted." Describe in more detail the circuitry provided to sense this condition. The discussion should include the method of maintaining independence between the two channels in view of the fact that a single switch (for each rod) is used to establish the "rod-in" condition.

7.7 Resolve the apparent inconsistencies between Section 7.16.4.2.2(1) and Figure 14.5-1 regarding the pressures at which the relief valves close and the number of groups into which the relief valves are divided in the REVAB system.

7.8 In previous BWR designs, a reactor trip function designed in accordance with IEEE Std 279 was provided for low condenser vacuum, either directly from sensing low vacuum or indirectly via closure of the main steam stop valves. Discuss the reasons for departing from this practice and describe the extent to which the proposed design, utilizing the turbine control system to provide an indirect reactor trip via closure of the turbine steam valves, meets the requirements of IEEE Std 279-1971.

7.9 Describe the means proposed to provide automatic indication in the main control room in the event that a protective action is bypassed or deliberately rendered inoperative. Provide the design criteria and bases for this indication and discuss the extent to which the indication can serve to:

(a) Supplement administrative controls regarding access to the means of bypassing and adherence to written test and maintenance procedures, and

(b) Aid the operator in assessing the availability of system level protective actions, i.e., recognizing the effects on plant safety of bypasses at the component level and of bypasses in auxiliary or supporting systems of ESF systems.

7.10 Appendix H discusses the Nine Mile Point 2 design conformance to certain items of concern documented in the ACRS letter dated December 20, 1967, on the Diablo Canyon Unit 1 plant. Expand this discussion to include the Committee's comment concerning the provision of continuous indication of the position of each control rod.

7.11 Describe in more detail the control design features provided to insure proper automatic and manual operation of the Standby Gas Treatment System. Your response should also address the instrumentation provided to the operator for monitoring the operation of this system.

7.12 Many of the Figures in Section 7.0 and throughout the PSAR do not contain information (explicit cross references, grid coordinates on the drawing, etc.) that is necessary to follow circuits, piping or logic from one figure to another. These Figures should be revised as required to enable the complete review of the systems they describe.

8.0 ELECTRIC POWER SYSTEMS

8.4 The information on the offsite power system design is not complete enough for evaluation of conformance with the requirements of General Design Criterion 17. Submit the following additional information:

(a) Provide a line diagram clearly showing all offsite power connections to Nine Mile Point Nuclear Unit 2, all such connections to Nine Mile Point Nuclear Unit 1 and the James A Fitzpatrick Plant, and all interconnections between these three plants.

(b) Provide a scaled plan layout of the site (including Unit 1 and the James A. Fitzpatrick Plant) showing all switchyards, transmission lines and associated rights-of-way, the location of switchgear and power transformers, and the routing of control and power circuits to remote structures. Identify all overhead and underground circuits. It is noted that this information request is partially satisfied by Figure 1.4-2. Correct the inconsistency between the description of the 115 kV transmission line arrangement given in Section 8.3.2.2 and that shown in Figure 1.4-2.

(c) Identify any transmission line crossovers, onsite or offsite, which could jeopardize the availability of offsite power. Verify that structural failure of any one line would not result in the failure of all other offsite power. Identify and justify any aspect of your design that does not meet the requirements of General Design Criterion 17.

(d) Section 8.14.3(16) states that "there are no electrical connections except communications between Nine Mile Point Nuclear Station Unit 1 or James A. Fitzpatrick Nuclear Power Plant." The meaning of this statement is not clear; correct it as required for consistency in accordance with your response to Item (a) above.

8.5 It is not clear from the information presented in PSAR Section 8.0 that your design of the power systems important to safety will meet the testability requirements of General Design Criteria 18. Describe the capability to be provided for periodically testing (a) the operability and functional performance of the offsite and onsite power sources and associated components, and (b) the operability of these electric power systems as a whole and under conditions as close to design as practical including the full

operational sequence that brings these systems into operation. This description should be sufficiently detailed to permit evaluation of conformance to all the requirements of General Design Criteria 18.

8.6 Justify the absence of any Safety Objective or Safety Design Bases for the reserve (offsite) a-c power systems (Section 8.4.2).

8.7 Correct or supplement your description of the Station Service a-c Power Distribution System (Section 8.5.4.1) as follows:

(a) It is stated that "transformers T5 and T6 will be fully rated to carry necessary loads on all 4,160 V switch groups during normal operation or during any postulated design basis accident." This seems to imply that load shedding may be necessary. Discuss this aspect of your design in more detail.

(b) The sixth paragraph appears to be a typographical error. Correct as necessary to make it meaningful.

(c) It is stated that "Each of the 4,160 V switch groups for essential auxiliaries can be supplied from the normal or reserve a-c power sources through either stepdown transformer T5 or T6, etc.". It appears from Figure 8.2-1 that this is not true. The present scheme allows only two of the three 4 kV normal buses to be energized from either T5 or T6. Neither transformer can be isolated from its normal 4 kV buses in order to permit these buses to be energized from the remaining transformer. Resolve this inconsistency.

(d) Identify the balance-of-plant loads that are supplied from the three standby 4,160 V switch groups (buses 2-101, 2-102, and 2-103) and from the three standby 600 V switch groups (buses 2-111, 2-112 and 2-113). Discuss that aspect of your design which will "disconnect these loads as required to ensure adequate power supply to all safety systems" under all operating and accident conditions.

8.8 The PSAR states (Sections 8.5.4.1, 8.7.3 and 8.8.3) that non-safety loads are being supplied from the redundant buses of the 4,160 V a-c standby system and the 250 V and 125 V d-c systems. What separation criteria are applied to the cables serving these non-safety loads in order to ensure the independence of the respective redundant power sources which supply them?

8.9 The criteria for preserving the independence of redundant safety cable and raceway systems (Section 8.5.4.2) should be supplemented as follows (your response to this question may be combined with your response to that portion of Question 7.1 which refers to Question 5 of Information Guide 2).

(a) The criteria should be provided for cable splices, including their location in the cable-raceway system. It is noted that in recent incidents faulty cable splices have been identified as the cause of fire in cable trays.

(b) The criteria do not specifically prohibit the routing of more than one redundant cable system through the same environmentally hazardous space. Correct this deficiency.

(c) The criteria do not address the ordinal arrangement by voltage class of parallel cable trays in a vertical tier, or the minimum separation between non-redundant cable trays in such a configuration. Correct this deficiency.

(d) The criteria do not include minimum separation or barrier requirements for cable trays containing redundant safety cables. Correct this deficiency. Your response should address all configurations used in your design, i.e., parallel tray arrangements, crossovers, etc.

(e) The criteria with regard to intermixing safety cable of different safety divisions in the same tray or conduit, etc. are not clear. Provide criteria that explicitly prohibit such intermixing of redundant safety cables.

(f) The criteria for cable tray loading appear to contain typographical errors. Correct these errors as necessary. Your response should also address (i) to the type and size of the cable trays used in your design, and (ii) to the possible failure of the insulation of the bottom layer of cable due to compaction over the design life of the plant.

(g) With regard to cable derating, the IPCEA-IEEE criteria used should be identified by title and publication number, and an explicit description of the application of the criteria should be provided.

(h) The criteria on overload protection are not clear. Revise these criteria as necessary to make them more meaningful.

(i) The criteria for the identification marking of safety related electrical components is not acceptable. We interpret Paragraph 4.22 of IEEE Std 279-1971 as requiring distinctive color coding of cables, cable trays, conduits, ducts, control cabinets, and other major electrical components including pump motors. This coding should differentiate between redundant safety channels or divisions and between safety and non-safety items. Submit revised criteria for implementing this requirement.

(j) Supplement the criteria for alphanumeric cable identification by providing typical examples of cable identifications which illustrate how your system differentiates between safety-related cable of different channels or divisions, non safety-related cable which is run in safety trays, and non safety-related cable which is not associated physically with any safety division.

8.10 In Sections 8.7.3 and 8.8.3 it is indicated that two breakers must be manually closed before two redundant power sources (batteries or chargers) can be paralleled on either the 250 V d-c or the 125 V d-c systems. Describe the alarms or indications that will be available in the control room to inform the operator of the status of these breakers and of the paralleled or non-paralleled status of the power sources.

8.11 Justify the absence of any Safety Objective or Safety Design Bases for the 24 V d-c system.

8.12 In Section 8.11.2(3), identify the portions of this system which do not meet the requirements of IEEE Std 308-1971.

8.13 The power generation design basis (Section 8.10.2[1]) for the "uninterruptable subsystem of the 120 V a-c normal power system" is not clear. Revise this section as necessary for consistency with Section 8.10.3(a).

8.14 Provide the following additional information on the three Diesel Generators (D/G) which comprise the onsite power sources for the standby a-c system.

(a) The overload ratings (2000 hour, 2 hour) of the D/G sets, reference Section 8.6.4.

(b) The cranking capacity of the D/G starting systems in terms of the number of consecutive starting attempts available, reference Section 8.6.3.

(c) It appears that the D/G protective trip circuits could be modified to reduce the likelihood of false trip during emergency conditions without unduly exposing the D/G units to destructive hazards. Justify the omission of such a trip bypass feature in your design. Also, further define the "exercise mode of operation during any design basis accident" referred in Section 8.6.3.

(d) The standby diesel generators must have the capability for starting and accepting design loads in the required time interval and sequence. This requirement should be reflected in the various descriptions of proposed D/G testing, reference Sections 8.6.6(4) and 8.14.1(c).

(e) The Acceptability Criteria in Section 8.14.1 should include the number of successful tests (and allowable failures) required to demonstrate acceptability, and in addition should include criteria for evaluation of the performance of the design loads under the voltage and frequency transients experienced during the tests.

8.15 The HPCS pump drive is described as a 3000 HP motor in Section 8.6.4. This motor is listed in Table 8.6-2 as 3050 HP and 2590 HP, resolve this inconsistency.

8.16 Provide the following information for all of the D.C. power systems.

(a) A layout sketch illustrating the physical location of each battery, separation of redundant batteries, and identification of each battery.

(b) An evaluation of the ability of the battery chargers to charge or prevent further discharge of the batteries while supplying d-c system loads during other than normal conditions. Sections 8.8.3 and 8.8.4 describe the capacity of the battery chargers in relation to the "normal d-c system load."

(c) The protection provided against overcharging.

12.0 STATION STRUCTURES AND SHIELDING

12.7 Clarify whether loads determined by the equations listed in Table 12.5-1 take into account the effects of thermal expansion of equipment that may be attached to the structures. If not, how are these loads taken into account.

12.8 Furnish diagrams to illustrate the short and long term (through 10^6 seconds) thermal gradients that will be established in the steel and concrete portions of the containment. On the basis of these gradients, indicate the manner in which the values of modulus of elasticity are modified in the thermal stress analysis to account for the creep effects.

12.9 The proposed maximum value of tangential shear (160 psi), as shown in Section 12.3.9, that can be resisted by concrete alone is not based on tests of specimens under tensile loads. Actually the reinforced concrete containment is subjected to a two-dimensional tension. Therefore, we believe that until a detailed study under realistic conditions is conducted a more conservative allowable stress, e.g., 40 psi, is appropriate. In this connection, indicate the reinforcing necessary to resist the tangential shear and describe the method of the analysis.

12.10 Provide a detailed sketch of the wall near the top of the drywell and the connection to the drywell cap. Indicate the manner in which the axial loads and the horizontal and vertical shears are carried by the wall at the support ledge for the drywell head.

12.11 To permit an evaluation of the reactor vessel supporting structure:

a. Discuss the manner in which the horizontal shears are carried by bolted or welded supports at the reactor skirt base.

b. List the allowable stresses which will be applied to the critical elements of the reactor supporting structure such as steel ring girder if any, bolts, and the concrete pedestal, under the combination of design loads. Provide a detailed sketch of the supporting structure.

12.12 For the primary shield wall:

a. Clarify the use of 1.2 S_y and 20% of the ultimate membrane unit strain and state the basis for using these allowables.

b. Provide the load combination equations.

c. Explain why seismic loads are omitted.

12.13 For the intermediate floor:

a. Indicate the structural testing that will be performed.

b. Describe what consideration has been given to the uplift in the floor due to downcomer jet reactions and vertical thermal growth of the reactor support pedestal.

12.14 State the magnitudes of the earthquake generated soil pressures (in terms of active and passive pressures) on exterior walls of Class I structures and indicate the method by which these pressures will be used to design the structural components.

12.15 Submit a list of computer programs that will be used in structural analyses to determine stresses and deformations of Seismic Class I structures. Include a brief description of each program and the extent of its application.

12.16 Describe the design control measures as required by 10 CFR Part 50 Appendix B that will be employed to demonstrate the applicability and validity of the above computer programs by any of the following criteria or procedures (or other equivalent procedures).

a. The computer program is a recognized program in the public domain, and has had sufficient history of use to justify its applicability and validity without further demonstration. The dated program version that will be used, the software or operating system, and the computer hardware configuration must be specified to be accepted by virtue of its history of use.

b. The computer program's solutions to a series of test problems, with accepted results, have been demonstrated to be substantially identical to those obtained by a similar, independently written program in the public domain. The test problems should be demonstrated to be similar to or with the range of applicability for the problems analyzed by the computer program to justify acceptance of the program.

c. The program's solutions to a series of test problems are substantially identical to those obtained by hand calculations or from accepted experimental test or analytical results published in technical literature. The test problems should be demonstrated to be similar to the problems analyzed to justify acceptance of the program.

12.17 Provide a summary comparison of the results obtained from each computer program with either the results derived from a similar program in the public domain, on a previously approved computer program or results from the test problems.

14.0 UNIT 2 SAFETY ANALYSIS

14.3 Provide a discussion of the consequences of control rod misalignment. If this is presented elsewhere in the PSAR, cross reference in Section 14.5.3.

C.0 EQUIPMENT DESIGN CRITERIA

C.12 In order to evaluate the adequacy of the Seismic Classification of fluid systems important to safety, identify those items in a lower classification from those in Safety Guide 29 and provide a discussion justifying your proposed classification.

C.13 In order to evaluate the adequacy of the industry codes and standards applied to water and steam-containing components of fluid systems important to safety, identify where there are differences which result in a lower classification level from those in Safety Guide 26 and provide a discussion justifying your proposed classification.