MAR 6 1973

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. John 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 additional information is required. The information requested is described in the enclosure and pertains to the different sections of the Preliminary Safety Analysis Report. The sections are the following: 2.0 Site, 3.0 Reactor, 4.0 Reactor Coolant System, 7.0 Control and Instrumentation, 8.0 Electrical Systems, 10.0 Auxiliary Systems, 12.0 Station Structures and Shielding, and 14.0 Unit 2 Safety Analysis.

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In order to maintain our licensing review schedule, we will need a completely adequate response to all enclosed questions by April 17, 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 to this project, the extent of the extension will most likely be greater than the delay in your response.

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DATE ▶						

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[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

Sincerely,

Original signed by
Robert A. Clark

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

Enclosure:
Request for Additional Information

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

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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

2.27 Figure 2.7-7 indicates that the foundation slabs for the Control Building and the Radwaste Building will be founded on engineered fill and/or on walls extending to bedrock. As the relationships of the structural foundations are not clear, furnish structural sections extending through the Control Room Building and the Radwaste Building.

2.28 Section 2.7.10 states that structural fill for Class I (Category 1) structures, pipelines, tanks and tunnels will be a selected earth fill material, compacted in thin lifts, using a controlled moisture content, to at least 95% Standard Proctor Test. Provide more detailed data regarding these fill materials such as:

- (a) grain size distribution
- (b) logs of borings drilled in borrow areas
- (c) compaction test results
- (d) dynamic test results on compacted samples, and analyses of response to dynamic loading
- (e) provide a summary of quality control specifications
- (f) provide representative laboratory test records to substantiate backfill characteristics.

2.29 A summary of the results of compression tests on rock samples is presented in Table IB-4. Provide the laboratory test records to substantiate these data.

2.30 Provide, if available, records of performance of foundations for Unit 1 and/or for FitzPatrick such as:

- (a) settlement records
- (b) discussion and evaluation of any structural deformation that could be attributed to differential settlement, heave, or uplift.

2.31 Evaluate the effects of hydrostatic uplift forces against structures and piping located at subsurface and below ground water level.

2.32 Page IC-4 describes "pop up" features which have either occurred or have been discovered within the past several years. These features are attributed to post-glacial crustal rebound and/or unloading horizons of rock containing internal stresses by removal of overburden. Define plans for detecting and evaluating such features should they occur during or following excavation for the nuclear power plant.

2.33 Your response to request 2.8 is unacceptable. Provide assurance that the local and interior drainage systems will effectively protect safety-related facilities, including the roofs and drainage of safety-related buildings, for the probable maximum precipitation, as indicated in Hydrometeorological Report No. 33 U. S. Weather Bureau (now NOAA) 1956, of a maximum 1-hour of 8.4 inches, 2-hour of 11.8 inches, 3-hour of 14.9 inches, 4-hour of 17.5 inches, 5-hour of 20.0 inches, and a maximum 6-hour of 22.20 inches. For instance, show the ability of site drainage, including the roofs of safety-related structures and exterior penetrations, to safely store or pass runoff resulting from the local probable maximum precipitation without a loss of function of the safety-related facilities.

2.34 Your response to Question 2.9 is considered inadequate. Normal procedures for verifying any mathematical model requires reproducing several historical events at several locations. This verification has not been indicated of your two-dimensional model. In our independent analyses of the probable maximum surge (stillwater level), elevation 254.0 USLS 1935 should be used as the stillwater basis of design. This is the same probable maximum surge level as required at the nearby James A. FitzPatrick Nuclear Power Plant. Provide verification of your two-dimensional model and perform analysis using elevation 254.0 USLS as the stillwater level.

2.35 Rather than assuming the solitary wave theory for breaking wave height is a function only of water depth as shown in response 2.11-1b., the most recent relations as developed by Weggel (ASCE Nov. 1972, Journal of the Waterways, Harbors and Coastal Engineering Division) should be used. These relations indicate that the maximum breaker height is a function of not only the water depth but wave period and bottom slope. Under certain conditions the wave height breaking on the structure can be equal to or greater than the water depth at the structure. Substantiate that the intake structure can withstand the force of waves breaking on the intake structure for the probable maximum surge, stillwater elevation 254.0 ft. USLS 1935 instead of a stillwater elevation of 252.7 ft. USLS 1935, and modify or amend response 2.11-2.

2.36 Recalculate the stability of the rubble dike structure based on the maximum breaker height as determined by the Weggel (1972) procedure and modify or amend response 2.11-3 accordingly. Independent calculations indicate design wave height should be in the order of 14.5 feet which would result in an armor stone weight of 13 tons (26,000 lbs).

2.37 Although we agree with wave runup on the dike of 15 feet with a 20-second wave period based on water level elevation 252.7 ft., waves with periods ranging from 8 to 20 seconds would produce runup greater than 15 feet. Independent calculations based on water level elevation 254.0 ft. (1 on 30 bottom slope in front of the dike and 1 on 2 dike slope) indicate the maximum runup would be in excess of 20 ft. for a wave period of 14 seconds. Overtopping would exceed 3 cfs per lineal foot of dike for waves with periods ranging from 12 to 20 seconds with a maximum overtopping in excess of 11 cfs for a H_b = 13.3 and T = 16 seconds. Your responses 2.11-4 and 2.12 indicate that the safe plant operation and maintenance of a shutdown mode does not depend on the integrity of the dike structure. Substantiate this conclusion based on the 254.0 ft. USLS 1935 water level without the dike, or evaluate the integrity of the dike to withstand the 14.5 ft. design wave and waves overtopping rates ranging from 3.0 to 11.0 cfs per lineal foot of dike, and provide assurance of the ability of the safety-related structures to function. Your calculations should be based on actual average bottom slopes and not a hypothetical slope.

2.38 Describe and furnish sufficient data, or make appropriate reference to pertinent sections in the PSAR, as indicated in Section 2.4.11.6 of the AEC report Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 1) to assure the requirements for the Ultimate Heat Sink are met. For guidance on ultimate heat sink, see Safety Guide 27.

3.0 REACTOR

3.5 What core average exposure (MWD/T) is used for the initial core fuel and the replacement core fuel as the design basis for the thermal and hydraulic core characteristics.

3.6 Describe the "getter" shown on Fig. 3.2-3 of the PSAR. Provide information on its performance during normal and abnormal conditions. Provide details of the fuel pellet dimensions that will be used for Nine Mile Point 2.

3.7 Your response to Question 3.4(a), which furnished revised scram characteristics, indicates that expected revisions in void and Doppler functions will significantly compensate the effects of the scram function, and that these revisions will not materially alter the analyses of abnormal transients for the detailed design. Provide additional information justifying this position. This may consist of reference, if applicable, to analyses performed on a power plant for which the revision to scram, void, and Doppler functions has been made.

3.8 Your response to Question 3.4(d) is inadequate concerning details of the gadolinia core design. Furnish details of what the preliminary core design is. Include a comparison of the possibilities for NMPNS with the final core designs of other power plants which are in various stages of review.

3.9 Provide justification that the number and location of LPRM and TIP detectors provided is adequate to furnish measurements ensuring that operating limits for linear power density (peaking factor) and CHFR are not exceeded. Provide information on errors present in these measurements, including evaluation of errors in view of allowable detector failures. Furnish detailed information on failure rates of the LPRM and TIP systems gained in operating reactors to date, to support assumed failure limits.

4.0 REACTOR COOLANT SYSTEM

4.18 The response to Question 4.12, which discusses the decoupler device for the pump-motor, needs clarification or additional information. The upper sprags that are used to prevent reverse rotation are not shown on Figure R4.12-1. Are these part of the original pump-motor assembly or part of the replacement decoupler?

4.19 The response to Question 4.14 does not provide analyses and details to show the provisions incorporated in the containment and Core Standby Cooling System (CSCS) suction lines to assure that a break in a pump suction line will not result in the loss of CSCS functions. Previous BWR plants have incorporated safeguards in the design to insure that a loss of CSCS function does not occur even if an uncontrolled leak occurs in an CSCS pump suction line. Provide information on the incorporated safeguards used for Nine Mile Point 2.

4.20 Provide analysis to assure that pressure drop in the suction lines (e.g. elevation, velocity, frictional) of the core standby cooling systems will not cause a line pressure below the vapor pressure of the liquid assuming the worst case, e.g. minimum containment pressure, maximum water temperature and minimum suppression pool level.

4.21 The section describing the RHR system alludes to a piping design for each loop that will preclude potential water hammer effects. Describe the system design including instrumentation, alarms, and any other provisions that monitor this system. Describe operator procedures needed to place the system in a standby condition. In addition, describe your systems for alleviating the water hammer effects in all core standby cooling pump systems.

7.0 CONTROL AND INSTRUMENTATION

7.14 The response to Question 7.1 (Item 11 of Information Guide 2) is not complete. Specific design criteria such as power supply redundancy and accuracy of the indicators were not addressed in all cases. The discussion of design adequacy should address the margin between the ranges of the indicators and recorders and the expected variations of the monitored parameters. Provide a complete response to Item 11 of Information Guide 2.

7.15 The response to Question 7.4 does not address Part a. of the question. PSAR Figure 7.3.11b (the revised figure supplied with Supplement 6) shows that either of two pressure switches (N018A or N018B) would prevent opening valves F008 and F009 if high pressure were sensed. However, PSAR Figure 7.3.10a (revised figure) is confusing because in the RHR line pressure switch (N018) and valves F008 and F009 have been removed. It also appears that the interlocks you proposed will be ineffective in preventing opening of the valves when the reactor pressure is high. Complete the answer to Question 7.4 and resolve the apparent inconsistencies between revised PSAR Figures 7.3.10a and 7.3.11b.

7.16 The response to Question 7.5 does not state that the differential pressure permissive interlocks are required for safety, nor does it address the effect of this interlock on the reliability of the LPCS and LPCI systems. Complete your response to Question 7.5 in this regard.

7.17 With reference to the response of Question 7.6.

(a) It appears that the statement on PSAR Page 7.5-1 regarding the "single-failure-proof circuitry" used to sense all rods inserted is incorrect. Resolve this apparent inconsistency.

(b) We have concluded that the testing described is a probable subject of the Technical Specification and should be included in PSAR Appendix B. State your intent in this regard.

(c) Resolve the apparent inconsistency between PSAR Section 7.5.3 which describes the safety objectives of the refueling interlocks and the response to Question 7.1 on Page R7.1-6 which indicates that the refueling interlocks are not required for safety.

7.18 Your response to Question 7.9 is not acceptable. The following comprises an acceptable method for implementing the requirement of Section 4.13 of IEEE Std 279-1971 and Criterion XIV of Appendix B to 10 CFR Part 50 with respect to indicating the bypass or inoperable status of portions of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety related functions:

(a) Administrative procedures should be supplemented by a system that automatically indicates at the system level the bypass or deliberately induced inoperability of the above safety-related systems.

(b) Automatic indication should be provided in the control room for each bypass or deliberately induced inoperable status that meets all of the following conditions:

   (i) Renders inoperable any redundant portion of the above-cited systems,

   *(ii) Is expected to occur more frequently than once per year, and

   *(iii) Is expected to occur when the affected system is normally required to be operable.

(c) Manual capability should exist in the control room to activate each system-level indicator.

(d) The design criteria for the indication system should reflect the importance of both providing accurate information for the operator and reducing the possibility for the indicating equipment to affect adversely the monitored safety systems. In developing the design criteria, the following should be considered:

   (i) The bypass indicators should be arranged to enable the operator to assess readily the operating status of each safety system and determine whether continued operation is permissible.

   (ii) When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.

   (iii) Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated causes of erroneous indications cannot be eliminated by another practical design.

   (iv) Unless the indication system is designed in conformance with criteria established for safety system, it should not be used to perform functions that are essential to the health and safety of the public. Neither should administrative procedures require immediate operator action based solely on the bypass indications.

* These conditions are interpreted to include bypasses that result from manipulation of permanently installed electrical control devices located on local and main control room panels.

(v) The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on the plant's safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment and the bypass indication should not reduce the required independence between redundant safety systems.

(iv) The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and/or annunciating function can be verified.

Discuss the degree of conformance of your design with the above criteria. If your design is based on different criteria, discuss your reasons for concluding that the design provides an equivalent degree of protection.

8.0 ELECTRICAL SYSTEMS

8.17 We interpret GDC 18 to require testability at power of the onsite emergency power sources (diesel generators) including the starting logic and load sequencing which is required to operate during emergency conditions. Supplement your response to Question 8.5 to include a description verifying the required testability or a commitment that this testability will be provided.

8.18 Supplement your response to Question 8.6 (c) as follows: Justify your retention of diesel generator trips for low cooling water pressure and bearing oil pressure during the DBA mode of operation. Your response should address (a) the time available for operator action (manual D/G trip) between occurrence of the trip condition and D/G failure assuming automatic trip was bypassed, and (b) the redundancy being provided to preclude false trip due to pressure switch failure.

8.19 Your response to Question 8.6 (d) regarding the non-applicability to D/G 2G-102 of the requirement for "starting and accepting design loads in the required time and sequence" is not correct, see Table 8.6-1. Revise your response to Question 8.6 (d) and Section 8.14.1 (c) accordingly.

8.20 Revise Table 8.6-2 to (a) indicate units where kW values are used (or change to corresponding HP values - see your response to Question 8.15), and (b) correct the HP and kW totals in the 2G-102 columns. Confirm that the remainder of the tabulated loads and totals are correct.

8.21 Your response to Question 8.9 is unacceptable in the following respects. Our comments are referenced to the applicable numbered sections of your response to Question 8.9:

(a) Section 4.2 and 5.1 Three separation divisions are required by your ESFS design as identified in Section 4.2. However, only two divisions are listed in Section 5.1 for the Reactor Building. Resolve this discrepancy.

(b) Section 5.2 Your criteria permit cables from more than one redundant safety division to be routed in the same cable tunnel. This is in violation of GDC 22 (single failure). Revise your criteria to preclude this arrangement or demonstrate that your design is acceptable on some other defined basis.

(c) Section 5.3 Provide separation criteria for cables of two redundant safety divisions which are routed in the same cable chase. The arrangement permitted by these criteria must meet the requirements of GDC 22.

(d) Section 6.2.3 Your criteria for separation of redundant safety cables and cable trays are unacceptable in the following respects. Revise your criteria to include the following. These recommendations are based on the criteria found acceptable in recently approved construction permit applications:

(i) Power and control cables will be installed in ladder-type trays and conduits.

(ii) Cables associated with redundant equipment shall be routed in separate conduits, cable trays, ducts, penetrations, etc.

(iii) As a minimum where vertical stacking of trays containing different classes of circuits in the same division in a horizontal run is necessary, a vertical spacing between trays of at least 12 inches is maintained between the top of the tray below and the bottom of the tray above. Where physical limitations prevent this, rigid conduit is used.

(iv) Three-foot horizontal and/or five-foot vertical separation shall be maintained between trays associated with redundant circuits regardless of voltage classification. Where such separation is impractical, enclosed steel tray or conduit will be used or fire barrier installed.

(v) Where vertical shafts are used between elevations, the same philosophy of separation is followed. In addition, all cable openings between elevations are sealed.

(vi) In tray crossovers, at least an 18-inch clear space separation will be maintained and enclosed tray will be used for 5 feet on each side of the crossover.

(vii) All cables entering the cable spreading room and control room areas and interconnecting cables between these two rooms are sealed with fire resistant material to assure the integrity of each area.

(e) Section 12 Supplement your response to include color coding of cable to differentiate between cables of redundant safety divisions, between safety and non-safety cables, and between non-safety cables which have been routed with cables of a safety division. Cables shall be marked in a manner of sufficient durability and at a sufficient number of points to facilitate initial verification by visual inspection that the cable installation is in conformance with the separation criteria. These cable markings shall be applied prior to or during installation.

(f) Section 12 Your alphanumeric cable identification scheme does not explicitly identify a non-safety cable which has been routed with cables of a safety division with which they are associated. These distinctions should be included in the alphanumeric designation. Revise your alphanumeric cable identification scheme to correct this deficiency.

10.0 AUXILIARY SYSTEMS

10.58 What are the maximum and average airborne radioactivity levels for normal operation, including anticipated operational occurrences, that will be allowed in areas within plant structures and within the restricted area where plant personnel or site visitors are permitted.

10.59 Provide a description of plant operating procedures to assure that onsite inhalation exposures will be kept as low as practicable during plant operation and maintenance.

10.60 Estimate the expected annual inhalation doses to plant personnel. Describe in detail the calculational methods.

10.61 The diesel generator rooms have been provided with two 50 percent capacity fans each. The response to Question 10.46 states that a failure of one fan will result in operating the diesel generator above rated ambient temperatures during summer design conditions and that the failure of both fans or a passive ducting failure will result in excessive room temperatures requiring switch over to the standby diesel generator. Provide the total time for operation before the diesel fails when continuously operating for both ambient conditions stated above, one fan and no fan, assuming no switch over.

12.0 STATION STRUCTURES AND SHIELDING

12.20 The response to Question 12.18 was not complete. Provide a general description of design criteria for the erection and dimensions of shield walls, for penetrating through shield walls, and for radiation levels acceptable in the plant areas to be occupied by personnel during normal operational and maintenance activities. Provide geometric and physical models. For a particular region of high potential personnel exposure, such as the demineralizer area, provide complete calculational details including radionuclide concentrations, absorption coefficients, build-up factors, shield wall thicknesses and physical dimensions. What are the bases for the selection of maximum radionuclide concentrations and occupancy factors?

12.21 The response to Question 12.8 was not complete. The method of modifying the values of modulus of elasticity of reinforcing steel was only given. Provide the corresponding information regarding concrete.

12.22 Describe the accident conditions considered in the design of structures other than the primary containment, e.g the Reactor Building which contains safety related equipment, and indicate the design criteria, the load combinations, and the methods used to design structural elements for such conditions (e.g pipe whip, jet inpingement, etc.). If accident conditions were not considered, justify your position.

14.0 UNIT 2 SAFETY ANALYSIS

14.10 For the anticipated transients, demonstrate that water level is controlled with sufficient margin so as to prevent liquid entering the steam lines.

14.11 Under accident conditions with no water in the vessel, how long would the LPCI system alone take to refill the core to the level of the jet pump suction? How much jet pump leakage is assumed, if any?

14.12 Analyze a LOCA condition involving an HPCS line break on the discharge side, upstream of the isolation valves. Use the Interim Policy Statement evaluation model including single failure of the LPCI-LPCS Diesel. Provide curves of a) peak fuel clad temperature for various rod groups, b) core flow, c) fuel channel inlet and outlet quality, d) heat transfer coefficients, e) reactor vessel water level, and f) minimum critical heat flux ratio as a function of time. Indicate the time that effective core cooling is initiated, the time the fuel channel becomes wetted based upon item 4 of Appendix A, Part 2 of the Interim Policy Statement, and the time the temperature transient is terminated. For the analysis performed above, discuss and justify the peaking factors used.

14.13 Provide analyses of the control rod drop accident in the Nine Mile Point 2 gadolina-urania core. Provide curves of control rod worth and peak fuel enthalpy versus moderator density and core power at beginning of life (reference Supplement 1 to NEDO-10527 if appropriate).and at the other critical times in core life (e.g. 6500 MWD/T). Describe the procedures and systems provided which will prevent selection or withdrawal of out-of-sequence control rods. Provide histograms showing peak fuel enthalpy versus number of rods following a control rod drop accident assuming that the dropped rod is a) the highest worth in-sequence rod and b) the highest worth out-of-sequence rod.