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SEP 25 1972

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. Thomas J. Brosnan
Vice President and 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 required information is described in the enclosure and pertains to the area of material problems.

In order to maintain our licensing review schedule, we will need a completely adequate response to all enclosed questions by November 27, 1972. 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 this 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 extent of the delay in your response.

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

Sincerely,

Original signed by:
Roger S. Boyd

Roger S. Boyd, Assistant Director
for Boiling Water Reactors
Directorate of Licensing

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

OFFICE ▷	L:GCR	L:GCR	L:BWR			
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DATE ▷	9/22/72	9/22/72	9/25/72			

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1972 -463 -015

REQUEST FOR ADDITIONAL INFORMATION

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION 2

DOCKET NO. 50-410

4.0 REACTOR COOLANT SYSTEM

4.1 In reference to ferritic materials (including welds) of the reactor pressure vessel beltline, provide a list of specifications which include requirements and any imposed limits on residual elements (reportable and nonreportable) which are intended to reduce sensitivity to irradiation embrittlement. Any additional or special requirements should be indicated.

4.2 To assure that ferritic materials of pressure-retaining components of the reactor coolant pressure boundary will exhibit adequate fracture toughness under system hydrostatic tests, provide the following information:

(a) Will the proposed operating limitations during hydrostatic tests of the reactor coolant system use as a guide Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Code Section III fracture toughness rules, 1972 summer addenda (Code Case 1514). If not justify your plans from deviating from this guide.

(b) Indicate the degree of compliance with the AEC proposed, "Reactor Vessel Material Surveillance Program Requirements" § 50.55a, Appendix H, July 3, 1971. State the degree of conformance particularly with respect to the following:

a. Number of capsules,

b. Number and type of specimens,

c. Withdrawal Schedule,

d. Retention of archive material.

4.3 Provide a description of the material inspection program that will be used to verify the non-susceptibility of unstabilized austenitic stainless steels to intergranular attack. If the procedures of ASTM A-262, Practice E are not employed, describe in detail and justify your proposed test procedures.

4.4 State your requirements for control of delta ferrite in austenitic stainless steel welds, to avoid microfissuring in welds, especially as regards filler materials, welding procedure qualification, and the methods for determining delta ferrite content of the completed welds.

4.5 In reference to your proposed maximum allowable unidentified leakage rate in the reactor coolant pressure boundary, which is greater than 1 gpm, provide the following information:

(a) The length of a through-wall crack that would leak at the rate of the proposed limit as a function of wall thickness.

(b) The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

(c) The mathematical model and data used in such analyses.

(d) Experimental data confirming validity of the analyses described in (a), (b) and (c).

4.6 Furnish the ratio of the proposed maximum allowable total leakage rate for the reactor coolant pressure boundary during normal operation to the normal capacity of the reactor coolant makeup system, and to the normal capacity of the containment water removal system.

4.7 To demonstrate compliance with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems," provide the following information:

(a) The acceptance standards that will be used to establish acceptability of the vessel for service during preoperational mapping of the reactor vessel by ultrasonic examination, to meet the requirements of IS-232 of Section XI of the ASME Code.

(b) Describe the access that will be provided which will permit inservice inspection of the Group B and C fluid systems such as the engineered safety systems, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems outside the limits of the reactor coolant boundary.

4.8 Provide a description of any special fabrication processes, new materials specifications, and nondestrictive examination techniques that will be employed in the construction of the reactor vessel.

4.9 Describe specific features incorporated in the reactor vessel that may contribute to improve service reliability and safety, (e.g., location of weld seams to facilitate inservice inspections, improvement of surface finishes to enhance nondestructive examinations, enhanced material physical properties, and augmented inspections over and above those required by Code rules).