NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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DOCKET NUMBER: 50-220

FILE NUMBER:

TO: MR. GEORGE LEAR

FROM: NIAGARA MOHAWK POWER CO. SYRACUSE, NEW YORK MR. GERALD K. RHODE

DATE OF DOCUMENT: 5/14/76

DATE RECEIVED: 5/18/76

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DESCRIPTION

LTR. W/ATTACHED RE OUR 4/12/76 LTR............ FURNISHING REQUESTED INFO. RE POTENTIAL CRACKING IN REACTOR FEEDWATER NOZZLE BLEND RADII.

PLANT NAME:

NINE MILE POINT #1

ENCLOSURE

ACKNOWLEDGED

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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

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SYRACUSE, N. Y. 13202

RECEIVED 5-18-76 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

May 14, 1976

Director of Nuclear Reactor Regulation
Attn: Mr. George Lear, Chief
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Lear:

Your letter dated April 12, 1976 requested information regarding potential cracking in reactor feedwater nozzle blend radii at Nine Mile Point Unit 1. The attached information answers the questions contained in your letter.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

GERALD K. RHODE
Vice President - Engineering

Regulatory Docket File

DOCKETED USNRC MAY 18 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

GKH/sz

Attachment

4967

Niagara·Mohawk Power Corporation
Information Concerning Potential Feedwater Nozzle Cracking
Nine Mile Point Unit 1

The following numbered items refer to questions contained in a letter from Mr. George Lear (Commission) to Mr. Gerald K. Rhode (Niagara Mohawk) dated April 12, 1976.

Question 1:

If inspection(s) of feedwater nozzle blend radii has (have) been accomplished, supply the following information:

(a) Date of inspection(s)

(b) Inspection method used

(c) Inspection results and subsequent actions - number of cracks found; number of cracks which penetrated the cladding into the base metal; maximum depth of cracks; general location of worst cracking around each nozzle; method used to remove cracks; cross-sectional area of reinforcement removed at the worst grind-out location; subsequent inspections and results.

(d) If inspection report(s) has (have) been previously submitted to the NRC containing the desired information of (a), (b), and (c), provide reference to the report(s) in lieu of the information in (a), (b), and (c).

Response:

(a) Ultrasonic examinations of reactor feedwater nozzles were performed on April 4, 1976 and April 9, 1976.

(b) Ultrasonic examination methods were used. The examination on April 4 employed a developmental technique using a 60 degree angle beam from the reactor vessel outer wall. All four feedwater nozzles were examined. On April 9, the southeast nozzle was re-examined, using a compound angle beam shear wave technique from the feedwater nozzle outside the vessel.

(c) The April 4 examination of the southeast feedwater nozzle revealed two indications at the clad-nozzle interface which were reportable in accordance with the test procedures. Five subsurface indications were found in the other three nozzles; these indications were interpreted as base metal inclusions. The April 9 re-examination of the southeast nozzle revealed no reportable indications.

(d) Not applicable.

Question 2:

If no inspection of feedwater nozzle blend radii has been conducted, provide date inspection is planned and date of next planned refueling outage.

Response:

Not applicable.

Question 3:

If inspection(s) and repair(s) has (have) been undertaken, provide date of next planned inspection.

Response:

The next inspection is planned during the Spring 1977 refueling outage.

Question 4:

Provide the number of startup/shutdown cycles to the date of the inspection, if already accomplished, and the number of cycles since such inspection. If no inspection has been conducted, provide the estimated number of cycles at the planned inspection date. A startup/shutdown cycle is defined as a power increase from zero and subsequent return to zero.

Response:

One-hundred (100) startup/shutdown cycles occurred from initial reactor operation to the ultrasonic examination performed on April 4. Subsequent to that examination, one and one-half (1 1/2) cycles have occurred.

Question 5:

Describe actions taken or planned to minimize cold, intermittent feedwater flow to the reactor pressure vessel. Include changes in operating procedures and any redesign of feedwater heaters, pumps, flow control valves, piping etc.

Response:

A principal cause of cold intermittent feedwater flow to the reactor vessel is the "jogging" of feedwater pumps during plant startups. This practice is not used at Nine Mile Point Unit 1. During reactor startups efforts are made to maximize the steaming rate. The various drains to the condenser hotwell are opened as early as possible. These practices maximize the rate of feedwater flow, thus avoiding rapid introduction of cold feedwater to the hot nozzles. During hot standby operation the low feedwater flow condition is avoided. Maintaining feedwater flow avoids potential thermal transients to the nozzles associated with increasing cold feedwater flow from the low flow condition.

Question 6:

Describe reactor pressure vessel pressure-temperature limits for operating conditions and especially for the inservice-hydrostatic and leak tests.

Response:

Reactor pressure vessel minimum temperature limits as a function of reactor internal pressure have been calculated for hydrostatic leak tests and power operation based on postulated feedwater nozzle flaws, as discussed below. Operating procedures are being developed to implement these limits, which are shown in Figure 1.

Minimum pressurization temperatures were calculated using the criteria of Appendix G of Section III of the ASME Boiler and Pressure Vessel Code, (1974 Edition, Winter 1975 Addenda) and the following conservative bases:

1. A postulated flaw size of 2-inches deep was assumed. This value is slightly greater than the 1/4 T reference flaw size recommended by the Code and is significantly greater than would be expected based on crack growth analyses and ultrasonic examinations.

2. The A336 nozzle forging material was assumed to have an RT_{NDT} of $+40^{o}F$. This value is the maximum value permitted by the vessel procurement specification and is greater than would be expected for this forging material, particularly in the region from the nozzle surface to the 1/4 T point -- the location of the postulated flaw.

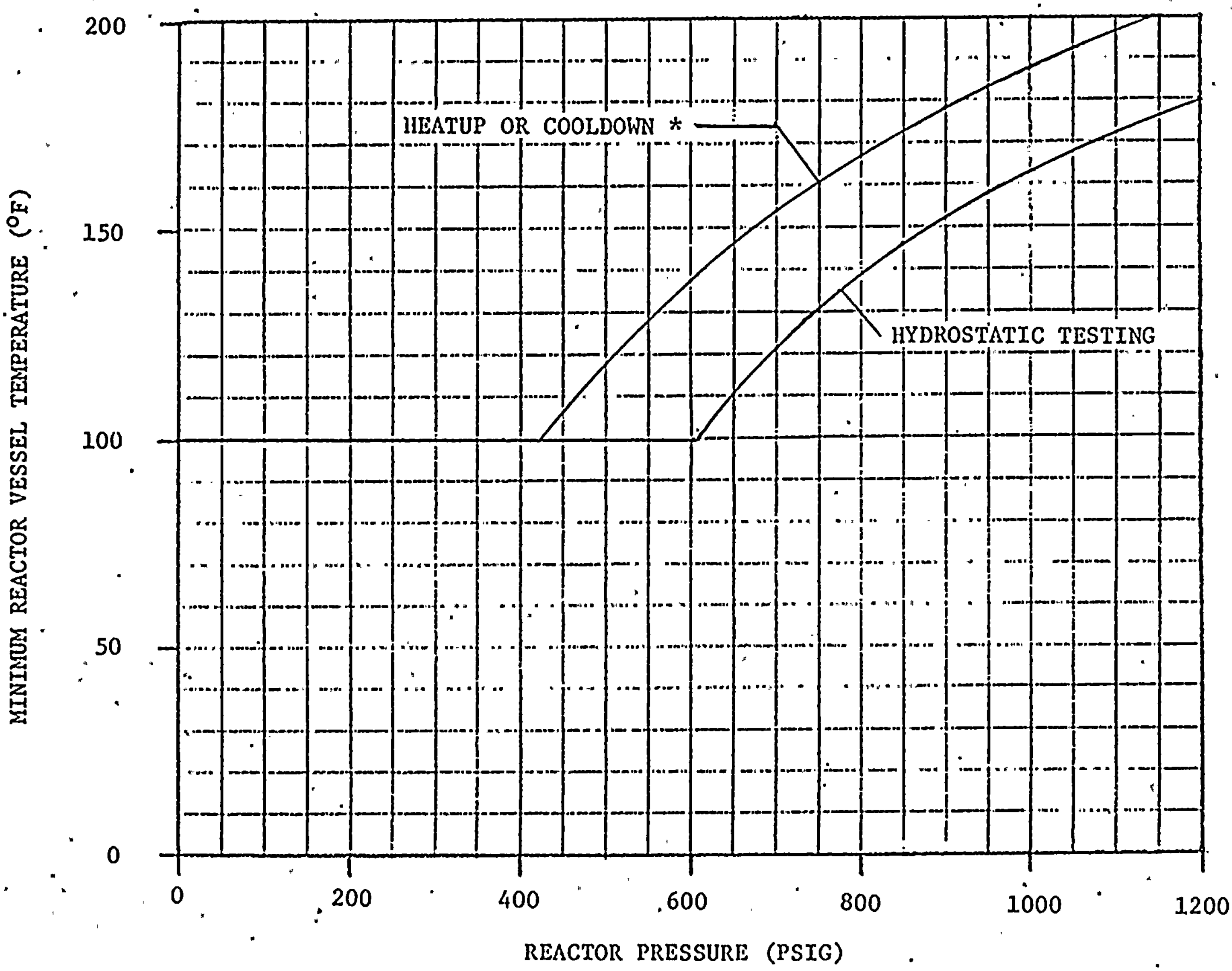
3. Pressure stresses and stress intensity factor, K_I, were determined using data presented in Welding Research Council (WRC) Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." Figure A5-1, for a 2:1 bi-axial stress field. (These data are more conservative than those recommended in WRC 175 for nozzle corner flaws.)

4. In applying the criteria of Appendix G of Section III, it was conservatively assumed that the total peak pressure stress is primary, while in fact, the total pressure stress includes secondary and peak stresses as well as primary stresses.

FIGURE 1

MINIMUM REACTOR VESSEL TEMPERATURE

FOR HYDROSTATIC TESTING AND HEATUP OR COOLDOWN



* Minimum temperature shall be 40 F higher when the core is critical.