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February 6, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactor Branch #1

Gentlemen:

DOCKET 50-266
ADDITIONAL INFORMATION INSERVICE INSPECTION PROGRAM
POINT BEACH NUCLEAR PLANT, UNIT 1

On May 20, 1977, we submitted a proposed inservice inspection and testing program for Point Beach Nuclear Plant, Unit 1, in accordance with 10 CFR 50, Section 50.55a. Following review of the program by the Commission Staff, a request for additional information was submitted to us with your letter of December 4, 1978.

As directed, we are enclosing three originals and forty copies of the responses to your requests. The responses are numbered to correspond with the requests.

Should you have additional questions concerning the information, please do not hesitate to call.

Very truly yours,

Sol Burstein

Executive Vice President

Sol Burstein

Enclosure

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DOCKET 50-266
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
INSERVICE INSPECTION AND TESTING PROGRAM
POINT BEACH NUCLEAR PLANT, UNIT 1

1a. Item B1.4, Examination Category B-D, Reactor Vessel Nozzle to Shell Welds and Nozzle Inside Radiused Sections

There are six nozzles in the reactor vessel; two inlet, two outlet and two safety injection. The original intent, as reflected in the technical specifications, was to examine the two outlet nozzles during the first inspection period, the safety injection nozzles during the second period, and the inlet nozzles during the third period. There is no access to the nozzle to vessel welds from the outside of the reactor vessel. These welds are examined from the inside using a reactor vessel inspection device (PaR Device). Using the PaR Device the core barrel must be removed to provide access to the inlet nozzles. Removal of the core barrel requires a complete unloading of all nuclear fuel from the reactor vessel. This is done only once during each 10-year inspection interval.

The outlet nozzles and nozzle to vessel welds were examined from the inside of the nozzles on schedule during the first inspection period. During the second inspection period it became necessary to remove the core barrel in order to inspect the vessel beltline welds. The safety injection nozzles were inspected during the second period because of the technical specification requirement. The inlet nozzles were inspected during the second period because the core barrel was removed.

The NRC was congnizant of, and concurred with, the early (compared with original intent) inspection of the inlet nozzles.

The reactor vessel inspections performed during the second period were performed in accordance with the 1974 code. The 1974 code increased the requirements for inspection of the nozzle to shell welds from those contained in previous codes. In order to better fulfill the increased requirements of the code and to provide a better test, the method of performing the test was changed from that employed during the first period. A "windmill" device was constructed for use on the PaR Device which enabled inspection of the welds by "scrubbing" the vessel walls in addition to inspecting from the bore of the nozzles. The "windmill" device and this method of inspection are only possible if the core barrel is removed. The outlet nozzles were reexamined using this method during the second period so that all six nozzle to vessel welds were examined during the second period. The reactor vessel inspections were performed during refueling 4, October 1976 to December 1976. It is anticipated that this pattern will be followed in the future such that all six of these welds will be inspected again during the second period of the next inspection interval.

No inspections were listed for the third period because the inspection interval requirements are complete. The better test method made possible by performing these examinations with the core barrel removed provides a positive effect on safety. Relief is requested from the requirements of paragraph IWB-2411.

1b. Item B1.12, Examination Category B-H, Reactor Vessel Integrally Welded Supports

There are two integrally welded reactor vessel supports which must be inspected. These vessel supports are not accessible from the outside of the reactor vessel. They are inspected from the inside of the vessel using the PaR Device. The core barrel and consequently the nuclear fuel must be removed from the reactor vessel in order to perform these inspections, therefore, it is not possible to split the inspections among different periods. These inspections were originally planned for the third period but were moved to the second period when it became necessary to remove the core barrel ahead of schedule as previously explained. These tests were performed during refueling 4. It is anticipated that they will be performed during the second period in succeeding intervals.

No inspections were listed for the third period because the inspection interval requirements are complete. Relief is requested from the requirements of paragraph IWB-2411.

1c. Item B1.14 B-I-1 Reactor Vessel Interior Clad Surfaces

Only a small amount of vessel cladding, less than seven inches, is visible above the core support structure during a normal refueling. At the time the cladding patches were examined it was felt that examination of locations below the core support ledge would be more meaningful than examinations in the normally visible area above the core support structure. Six cladding patches all below the core support ledge were examined during the second period during refueling 4. If desired, the cladding patches can be relocated to the normally visible area above the core support structure for the next interval. This would provide less meaningful but more time spaced examinations. If no adverse comment is received, it is anticipated that the same patches previously examined would be examined again during the second period of succeeding intervals, at least until the 1977 Code becomes applicable. The 1977 Code deletes inspection of these cladding patches.

No examinations were listed for the third period because the inspection interval requirements are complete. The more meaningful location of the cladding patches has a beneficial effect on safety. Relief is requested from the requirements of paragraph IWB-2411.

1d. <u>Item B4.6 Examination Category B-J. Branch Pipe Connection Welds Exceeding Six Inches Diameter</u>

There are four Class 1 branch connection welds exceeding six inches nominal pipe diameter in Point Beach, Unit 1. One of these four is required to be inspected during the ten-year inspection interval. It is not a requirement of the code that one branch connection weld examination be divided into three separate examinations. Two of the branch connection welds exceeding six inches in diameter were examined during the first two inspection periods.

No examinations were listed for the third period because the inspection interval requirements are complete.

1e. Item B6.5 Examination Category B-K-2 Valve Support Components

There are no valve support components associated with the Class 1 valves at Point Beach Nuclear Plant. The comment "Done per routine maintenance procedure" in our letter of October 6, 1977, is withdrawn.

2. Item B1.6, Examination Category B-F, Reactor Vessel Nozzle to Safe End Welds

There are six reactor vessel nozzle to safe end welds. The original intent was to perform examinations of these welds in conjunction with other examinations associated with the individual nozzles. That is, examine the welds on the outlet nozzles during the first period, those on the safety injection nozzles during the second period, and those on the inlet nozzles during the third. The outlet nozzle welds were examined on schedule during the first period and the safety injection nozzle welds on schedule during the second. When it became necessary to remove the core barrel ahead of schedule during the second period, examination of the inlet nozzles was also performed ahead of schedule during the second period. At the time the NRC was cognizant of and concurred with the early examination of the inlet nozzle welds. In addition, the outlet nozzle welds were reexamined with the result that all six nozzle welds were examined during refueling 4. It is anticipated that this pattern will be followed in the future such that all six of these welds will be inspected again during the second period of the next inspection interval.

No inspections were listed for the third period because the inspection interval requirements are complete. In fact, the requirements were exceeded because of the dual examination of the outlet nozzle to safe end welds. These additional examinations have a beneficial effect on safety. Relief is requested from the requirements of paragraph IWB-2411.

3. Item B3.1, Examination Category B-B, Steam Generator Tube Sheet to Head Circumferential Welds, and Item 3.2, Examination Category B-D, Steam Generator Nozzle Inside Radiused Sections

It is not our interpretation of the Code that the requirements of paragraph IWB-2411 coupled with Items B3.1 and B3.2 are to be applied separately to each steam generator.

There are two Class 1 steam generator tube sheet to head welds, one on each of two steam generators. The Code requires that 5% of each weld be inspected during the inspection interval. Approximately 8.8% of the A steam generator weld was examined during the first period and approximately 8.8% of the B steam generator weld was examined during the second period during refueling 4. It is not the intent of the code to require multiple inspections of a weld. The interpretation that the Code requires 1 2/3% of each weld to be inspected each of three times during an inspection interval is a distortion of the words of the Code and would result in additional and unnecessary radiation exposure to inspection and support personnel. The commitment in the test plan to examine 1 2/3% of this weld in the A steam generator is withdrawn.

No examinations were listed for the third period because the inspection interval requirements are complete. In fact, the requirements were exceeded by approximately 78%. The additional weld length examined above that required by the Code has a beneficial effect on safety. Relief is requested from the requirements of paragraph IWB-2411.

There are four Class 1 steam generator nozzle inside radiused sections in the plant, two on each of two steam generators. The two inside radiused sections of the B steam generator nozzles were examined during the second period and the two inside radiused sections of the A steam generator will be examined during the third period.

4. The stated argument in the NRC letter appears to be a misinterpretation of background information previously presented. We concur with the NRC Staff interpretation that currently classified Class 2 safety related systems should be inspected in accordance with the requirements of Section XI to the extent practical. Each safety related safety class system was carefully reviewed in detail to determine the applicable inspection requirements irrespective of time of classification. The review of the Class 2 safety related systems resulted in the determination that each may be exempted from the examination requirements of Table IWC-2520 in accordance with the criteria given in paragraph IWC-1220. Additional sampling points have been added to the piping systems and the sampling program has been increased in a number of areas to meet the system sampling

requirements. The Class 2 safety related systems will all be subjected to system pressure tests and will be inspected in conjunction with the system pressure tests as required by Section XI and as shown in the test plan.