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POINT BEACH NUCLEAR PLANT
UNITS 1 AND 2

INSERVICE INSPECTION PROGRAM
TECHNICAL EVALUATION REPORT

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U.S. Nuclear Regulatory Commission

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ATTACHMENT 1

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INSERVICE INSPECTION PROGRAM

INTRODUCTION

The revision to 10 CFR 50.55a, published in February 1976, required that Inservice Inspection (ISI) Programs be updated to meet the requirements (to the extent practical) of the Edition and Addenda of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code* incorporated in the Regulation by reference in paragraph (b). This updating of the programs was required to be done every forty months to reflect the new requirements of the later editions of Section XI.

As specified in the February 1976 revision, for plants with Operating Licenses issued prior to March 1, 1976, the Regulations became effective after September 1, 1976 at the start of the next regular 40-month inspection period. The initial inservice examinations conducted during the first 40-month period were to comply with the requirements in editions of Section XI and addenda in effect no more than six months prior to the date of start of facility commercial operation.

The Regulation recognized that the requirements of the later editions and addenda of the Section XI might not be practical to implement at facilities because of limitations of design, geometry, and materials of construction of components and systems. It therefore permitted determinations of impractical examination or testing requirements to be evaluated. Relief from these requirements could be granted provided health and safety of the public were not endangered giving due consideration of the burden placed on the licensee if the requirements were imposed. This report provides evaluations of the various requests for relief by the licensee of the Point Beach Units 1 and 2. It deals only with inservice examinations of components and with system pressure tests. Inservice tests of pumps and valves (IST programs) are being evaluated separately.

The revision to 10 CFR 50.55a, effective November 1, 1979, modified the time interval for updating ISI programs and incorporated by reference a later edition and addenda of Section XI. The updating intervals were extended from 40 months to 120 months in order to be consistent with intervals as defined in Section XI.

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* Hereinafter referred to as Section XI.

For plants with Operating Licenses issued prior to March 1, 1976, the provisions of the November 1, 1979 revision are effective after September 1, 1976 at the start of the next one-third of the 120-month interval. During the one-third of an interval and throughout the remainder of the interval, inservice examinations shall comply with the latest edition and addenda of Section XI, incorporated by reference in the Regulation, on the date 12 months prior to the start of that one-third of an interval. For Point Beach Units 1 and 2, the ISI program, and the relief requests evaluated in this report, cover the last 40 months of the current 120-month inspection interval, i.e., from August 1977 to December 1981* for Unit 1 and June 1979 to September 1982 for Unit 2. These programs were based upon the 1974 Edition of Section XI of the ASME Boiler and Pressure Vessel Code with Addenda through the Summer of 1975.

The two reactor facilities, Point Beach Unit 1 and Unit 2, are essentially identical. The main differences in their ISI programs are that some of the items in Unit 1 scheduled for examination during the third forty-month period were scheduled for an earlier period in Unit 2 and vice versa.

The November 1979 revision of the Regulation also provides that ISI programs may meet the requirements of subsequent code editions and addenda, incorporated by reference in paragraph (b) and subject to Commission approval. Portions of such editions or addenda may be used provided that all related requirements of the respective editions or addenda are met. These instances are addressed on a case-by-case basis in the body of this report.

References (1) to (22) listed at the end of this report pertain to information transmittals on the Inservice Inspection (ISI) Reports on Units 1 and 2 between the licensee, Wisconsin Electric Power Company, and the Nuclear Regulatory Commission (NRC). By letters of April 26 and November 22, 1976[(1)(3)], the NRC provided general ISI guidance. Technical Specifications changes in response to that guidance were made by the licensee on February 17, 1977 for Unit 1.[(4)] ISI program submittals were made on May 20, 1977[(5)] for Unit 1 and on February 26, 1979[(13)] for Unit 2. The NRC granted interim relief to Unit 1 on August 26, 1977[(7)]. By letters of August 3, 1977 and December 4, 1978[(6)(11)] the NRC requested additional information to complete the report for Unit 1. This information was provided by the licensee on October 6.

*The Point Beach Unit 1 first ten-year interval was extended to December 1981 to permit inspections to be concurrent with plant outages as allowed in Article IWA-2400 of Section XI.

1977 and February 6, 1979.[8][12] By letters of October 4, 1979 and March 12, 1982[14][21] the NRC requested additional information on both Units 1 and 2. This information was provided by the licensee on December 14, 1979 and April 14, 1982.[15][22] In addition the licensee submitted a 10-year ISI completion report on Unit 1 on February 23, 1982.[20]

From these submittals, a total of 7 requests for relief from code requirements or updating to a later code were identified. These requests are evaluated in the following sections of this report.

I. CLASS 1 COMPONENTS

A. Reactor Vessel

1. Nozzle-to-Shell Welds and Nozzle Inside Radiused Sections (Applies to Unit 1), Category B-D, Item B1.4

Code Requirement

A volumetric analysis of these welds shall be made according to the schedule given in paragraph IWB-2411, which states, "at least 25% of the required examinations shall have been completed by the expiration of one-third of the inspection interval (with credit for no more than 33-1/3% if additional examinations are completed) and at least 50% shall have been completed by the expiration of two-thirds of the inspection interval (with credit for no more than 66-2/3%). The remaining required examinations shall be completed by the end of the inspection interval."

Code Relief Requested

Relief is requested from the schedule given in IWB-2411.

Proposed Alternative Examination

All nozzles will be examined once every 10 years when the core barrel is removed.

Licensee's Basis for Requesting Relief

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 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 better test method made possible by performing these examinations with the core barrel removed provides a positive effect on safety.

Evaluation

The schedule for examining welds in the nozzle-to-shell welds and nozzle inside radiused sections for Point Beach Unit 1 originally was as follows:

1st period (40 months) -- 2 Outlet nozzles

2nd period (40 months) -- 2 Safety nozzles

3rd period (40 months) -- 2 Inlet nozzles

(core barrel must be removed).

During the second period it was necessary to remove the core barrel and all six nozzles were inspected using the "windmill" device. Removing the core barrel and reexamining the two inlet nozzles merely to comply with the schedule clearly is not practical from the standpoint of the personnel exposure incurred for only a marginal gain in safety. The total quantity of welds examined during the interval exceeds the requirements since the outlet nozzles were examined twice.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief from IWB-2411 should be granted and the proposed alternative of examining all six nozzles at one time when the core barrel is removed should be approved.

References

Reference 11, pg 1; reference 12, pp 1 and 2.

2. Safety Injection Nozzle to Safe End Welds (Applies to Unit No. 1), Category B-F, Item B1.6

Code Requirement

Volumetric and surface examination covering the circumference of 100% of the welds during each inspection interval. Examinations in each 40 month period shall be in accordance with paragraph IWB-2411.

Code Relief Requested

Request relief from the surface examination and from the requirements of IWB-2411.

Proposed Alternative Examination

None

Licensee's Basis for Requesting Relief

This inspection is not practical because these welds are not accessible from the outside. These welds were previously examined ultrasonically from the bore.

Evaluation

There are six nozzles in the reactor vessel; two inlet, two outlet and two safety injection. A surface dye penetrant examination is performed on all of the reactor nozzle safe-end connections with the exception of the two safety injection nozzle connections. Neither a visual nor surface examination can be performed on these welds since they are enclosed by a concrete sleeve. An ultrasonic examination of the welds of all six nozzles was performed from the inside diameter of the nozzles during the second period with the aid of a remote control examination device (PaR device) along with the nozzle to shell weld examination.

The once every 10-year volumetric examination provides sufficient information as to the condition of the safety injection safe-end weld. Surface examinations are made on the other four safe-end welds in accordance with the schedule given in IWB-2411, which does not necessarily match the timing of the once every 10-year volumetric examination. Should any of these surface examinations indicate a surface flaw, investigation should include an additional volumetric examination of not only the problem weld but also the two safety injection nozzle welds. If weld cracking should be more general than one specific weld, this investigation would provide timely information on the condition of the safety injection welds.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief from the surface examination of the safety injection safe-end welds should be granted. However, if the surface examination of any of the other reactor safe-end welds should indicate a surface flaw, an additional volumetric examination of not just the problem weld, but also of the two safety injection welds should be done during the shutdown.

References

Reference 5; reference 6, pg 3; reference 8, pg 13; reference 11, pg 1; reference 12, pp 1 and 2.

3. Integrally Welded Supports (Applies to Unit No. 1), Category B-H, Item B1.12

Code Requirement

A volumetric examination of a 100% of the welding shall be done according to a schedule given in paragraph IWB-2411.

Code Relief Requested

Code relief is requested from the schedule given in IWB-2411.

Proposed Alternative Examination

The volumetric examination of the reactor vessel welded supports would be done at once every 10 years when the core barrel is removed.

Licensee's Basis for Requesting Relief

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 practical 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. These tests were performed during refueling outage #4. It is anticipated that they will be performed during the second period in succeeding intervals.

Evaluation

The integrally-welded reactor vessel supports cannot be examined from outside the reactor vessel, and must be remotely examined from inside the reactor using the PaR Device. This examination can only be performed when the core barrel is removed. This fact makes compliance with IWB-2411 impractical.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief from IWB-2411 should be granted. The alternative schedule of examining all the integrally-welded reactor vessel supports at least once every ten years should be approved.

References

Reference 11, pg 1; reference 12, pg 3.

4. Vessel Cladding (Applies to Unit 1), Category B-I-1, Item B1.14

Code Requirement

A visual examination of six patches (each 36 sq. in.) evenly distributed in the accessible section of the vessel shell shall be made according to paragraph IWB-2411.

Code Relief Requested

Relief is requested from the schedule given in IWB-2411.

Proposed Alternative Examination

The examination on all of the patches would be done at one time when the core barrel is removed.

Licensee's Basis for Requesting Relief

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.

Evaluation

To comply with the cladding examination requirements would require removal of fuel and the core barrel. This is an impractical requirement with a relatively small compensation in safety. The examination that was completed at Point Beach Unit 1 during the second period is entirely adequate. Examination of nozzle-to-vessel welds covers sufficient cladding in more suspent areas of the vessel to determine the cladding condition.

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

(a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));

(b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used;

(c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

The requirements for examining closure-head cladding and vessel cladding are deleted from the 1977 Edition with addenda through Summer 1978.

Recommendations

Pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the Summer 1973 Addenda for Category B-1-1 items. This approval would delete the requirement to examine these items.

References

Reference 11, pg 1; reference 12, pp 3 and 4.

B. Pressurizer

No relief requests.

C. Heat Exchangers and Steam Generators

1. Regenerative Heat Exchanger, Integrally Welded Supports (Units 1 and 2), Category B-H, Item B3.7

Code Requirement

Volumetric examination of 10% of the circumference of the weld to the vessel during each inspection interval.

Code Relief Requested

Request relief from making a volumetric (ultrasonic) examination of the weld.

Proposed Alternative Examination

Visual examination of the accessible portion of the welds.

Licensee's Basis for Requesting Relief

Ultrasonic examination of the support-to-vessel tack welds is not practical because of the curvatures of the vessel end caps. Liquid penetrant examination of these welds is not practical due to masking caused by penetrant entrapment between the support member and vessel shell. Radiation levels around the residual heat exchanger are 2R to 3R.

Evaluation

The regenerative heat exchanger welded supports are three 1" long partial-length fillet welds between the heat exchanger and the saddles. This weld configuration does not lend itself to ultrasonic examination. Dye penetrant surface examination is not practical as an alternative examination, since the penetrant would be trapped in areas adjacent to the weld.

When the developer is applied relatively large amounts of penetrant would flow out of the adjacent areas and overshadow any penetrant which might be present as a surface flaw. A visual examination is the only practical method. The examination period will have to be relatively short since the radiation levels in the area are about 2-3R.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief should be granted from the volumetric examination provided the proposed alternative visual examination is performed.

References

Reference 5; reference 6, pg 4; reference 8, pg 15; reference 13, pg 2.7 .

D. Piping Pressure Boundary

No relief requests.

E. Pump Pressure Boundary

1. Reactor Coolant Pumps, Integrally Welded Supports (Applies to both Units 1 and 2), Category B-K-1, Item B5.4

Code Requirement

A volumetric examination of 25% of integrally-welded supports each inspection interval.

Code Relief Requested

Request relief from making a volumetric examination (ultrasonic) of the integrally welded support.

Proposed Alternative Examination

Visual examination.

Licensee's Basis for Requesting Relief

Volumetric examination of these welds is not practical. The surface is rough and ultrasonic waves do not propagate well in the cast stainless material.

Evaluation

The ultrasonic examination of the reactor coolant pump support lugs is not practical because the ultrasonic waves do not propagate well through the rough cast stainless steel material. A dye penetrant examination as an alternative inspection is also not practical because of the rough surface. After the penetrant has been allowed to dwell, it cannot be cleaned properly. When the developer is applied to the examination area, the trapped penetrant will be drawn out. Numerous false indications will appear, and any flaw indications will be indistinguishable. A visual examination appears to be the only suitable alternative examination.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief should be granted from the volumetric examination provided the proposed alternative of visual examination is performed.

References

Reference 5; reference 6, pg 3; reference 8, pg 14; reference 13, pg 2.11 .

2. Reactor Coolant Pumps, Pump Casing Welds (Applies to both Units 1 and 2), Category B-L-1, Item B5.6

Code Requirement

A volumetric examination of the weld metal and base metal for one wall thickness beyond the edge of the weld. The examination performed during each inspection interval shall include 100% of the pressure-retaining weld in at least one pump of each group. The examination may be performed at or near the end of the inspection interval.

Code Relief Requested

Request relief from making an examination of one full wall thickness beyond the edge of the weld on each side.

Proposed Alternative Examination

Examine the weld for a distance of 1/2-inch on either side of the weld per 1977 Edition, through Summer 1978 Addenda.

Licensee's Basis for Requesting Relief

For the reactor coolant pump casing weld examination, examining one wall thickness on either side of the weld would mean that a 17 to 20-inch band would have to be inspected for each weld. The technique and equipment which has just recently been developed to perform an inspection of the reactor coolant pump casing welds is not capable of examining bands of this width. Multiple shots using the miniature linear accelerator (MINAC) would have to be taken to cover these band widths. Thus, instead of about 35 shots per weld, about 100 shots for each of the three welds would have to be taken. Each shot requires an exposure time of one-half to three hours. There would be a substantial increase in the accumulated radiation exposure associated with the placement of the radiographic film and a substantial increase in the examination time with no additional benefits in flaw detection. As recognized by the 1977 Edition, an examination of 1/2-inch of the base metal on either side of the weld encompasses the expected flaw zone for the reactor coolant pump casing welds.

In this submittal, the licensee also described the results of the MINAC examination(s) to date and future plans as follows:

In accordance with the inspection items B5.6 and B5.7, radiographic examination of the Unit 1 "B" RCP casing welds and a visual examination of the pump inside pressure-retaining surface using the MINAC and manipulator was performed during the Unit 1 1981 refueling outage. Essentially, 100% of all the casing welds were examined. The only areas not radiographed were the areas under the pump support lugs and inaccessible portions of the discharge nozzle. The MINAC was first utilized at the Ginna Plant. In MINAC examinations performed at Point Beach, Turkey Point, and Ginna, no notable indications were found in any of the pumps examined.

The casing examination at Point Beach took about 25 days to perform, including the associated pump disassembly and reassembly, and resulted in a total accumulated radiation exposure of 36 man-rem and a cost on the order of $700,000. Prior to performing the examination on one of the Unit 2 reactor coolant pumps which are identical to those of Unit 1, an evaluation of the improvements in the inspection methods employed will be performed to determine if the total cost in outage time, exposure, and money can be reduced to a level more commensurate with the benefits of the examination. Current plans are to disassemble a Unit 2 RCP and perform the casing weld examination during the 1983 refueling outage, or a waiver will be requested after the results from the H. B. Robinson examination (being performed in April 1982) are available.

Evaluation

The examination required by the 1974 Code is not practical. No utility has done an inservice inspection of the reactor coolant pump casing welds in accordance with the requirements of the 1974 Edition of ASME Section XI. The diffuser of the pump makes it impossible to examine one weld thickness above the upper weld and the changing casing thickness, coupled with the physically allowed beam angles, makes examining one wall thickness on either side of any of the welds difficult.

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

(a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));

(b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used;

(c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

The MINAC examination of the Unit 1 pump casing weld complied with the more recent code. A Unit 2 pump casing weld will also be examined according to the more recent procedure, or else a waiver will be requested, later.

Recommendations

Pursuant to 10 CFR 50.55a(g)(4)(iv), approval to update to the 1977 Edition (through S-78 Addenda) should be granted. This would require only one-half-inch of base metal on each of the weld to be examined. The examination completed on Unit 1 would then be in compliance, as would the Unit 2 examination now planned for 1983.

References

Reference 5; Reference 13, pg 2.12; Reference 18.

F. Valve Pressure Boundary

(No relief requests).

II. CLASS 2 COMPONENTS

(No relief requests).

III. CLASS 3 COMPONENTS

(No relief requests).

IV. PRESSURE TESTS

(No relief requests).

V. GENERAL

(No relief requests).

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2. Sol Burstein (WE) to Bernard C. Rusche (NRC), Point Beach Nuclear Plant Use of Code Case 1968 for In-Service Inspection, June 2, 1976.

3. George Lear (NRC) to Sol Burstein (WE), November 22, 1976.

4. Sol Burstein (WE) to Bernard C. Rusche (NRC), Inservice Inspection of Safety Class Components, Technical Specification Change Request No. 42, Point Beach Nuclear Plant Unit 1, February 17, 1977.

5. Sol Burstein (WE) to Edson G. Case (NRC), Docket No. 50-226, Point Beach Nuclear Plant Unit 1, Inservice Inspection Program for Safety Class Components, May 20, 1977.

6. George Lear (NRC) to Sol Burstein (WE), August 3, 1977.

7. George Lear (NRC) to Sol Burstein (WE), August 26, 1977.

8. Sol Burstein (WE) to George Lear (NRC), Docket 50-266, Additional Information Inservice Inspection Program, Point Beach Nuclear Plant Unit 1, October 6, 1977.

9. A. Schwencer (NRC) to Sol Burstein (WE), July 12, 1978.

10. Sol Burstein (WE) to Harold R. Denton (NRC), Docket No. 50-301, Inservice Inspection of Safety Class Components Technical Specification Change Request No. 58, Point Beach Nuclear Plant Unit 2, November 27, 1978.

11. A. Schwencer (NRC) to Sol Burstein (WE), December 4, 1978.

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18. C. W. Fay (WE) to H. R. Denton (NRC), Docket Nos. 50-266 and 50-301, Inservice Inspection of Safety Class Components Updating of 10-year Plans, Point Beach Nuclear Plant Units 1 and 2, October 6, 1981.

19. Timothy G. Colburn (NRC) to Sol Burstein (WE), October 21, 1981.

20. C. W. Fay (WE) to H. R. Denton (NRC), Docket No. 50-266, Completion of First Ten-Year Inservice Inspection Interval, Point Beach Nuclear Plant, Unit 1, February 23, 1982.

21. Robert A. Clark (NRC) to C. W. Fay (WE), March 12, 1982.

22. C. W. Fay (WE) to H. R. Denton (NRC), Additional Information on Inservice Inspection Program, Reactor Coolant Pump, Point Beach Nuclear Plant, Units 1 and 2, April 14, 1982.