POWER AUTHORITY OF THE STATE OF NEW YORK

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September 28, 1982
JPN-82-75

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Inservice Inspection Program
Request for Additional Information

References: 1) P.J Early (PASNY) to T.A. Ippolito (USNRC) dated September 10, 1979 regarding Inservice Inspection Program - Inservice Examination of Welds and Supports (JPN-79-57)

2) D.B. Vassallo (USNRC) to L.W. Sinclair (PASNY) dated April 14, 1982 - same subject

3) J.P. Bayne (PASNY) to D.B. Vassallo (USNRC) dated June 21, 1982 - same subject (JPN-82-52)

Dear Sir:

The Power Authority submitted, via Reference 1, inservice inspection program relief requests for our FitzPatrick nuclear power facility. As a result of your initial review, you requested additional information on these relief requests via Reference 2. We found it necessary, for the reasons detailed in Reference 3, to reschedule the submittal for that additional information.

Through discussion with members of your staff, we clarified some of your questions. As a result, attached is a partial response to Reference 2.

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Items 8 and 10 require further research or study by the Authority and therefore are not addressed in the attachment to this letter. Specifically, some of this information can only be obtained when the plant is shut down. Other information requires the completion of an industry survey. A response to the remaining two questions will be provided by December 8, 1982 or approximately 30 days after the start of the next scheduled outage.

Also discussed with your staff was the schedule for the formal granting of these relief requests. We are proceeding on the basis that these relief requests will be granted upon identification of those specific components not inspected (per our response to items II.1-II.6), at the end of the ten year inspection interval.

If you have any questions, please contact Mr. J.A. Gray, Jr. of my staff.

Very truly yours,

J.P. Bayne
Executive Vice President
Nuclear Generation

cc: Mr. Ron Barton
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Attachment to JPN-82-75

Power Authority of the State of New York
James A. FitzPatrick Nuclear Power Plant

RESPONSE TO APRIL 14, 1982 REQUEST FOR ADDITIONAL INFORMATION - INSERVICE INSPECTION PROGRAM

These responses are keyed to the enclosure of the April 14, 1982 letter, D.B. Vassallo (USNRC) to L.W. Sinclair regarding the James A. FitzPatrick Inservice Inspection Program.

I. General

1. A relief request will be submitted prior to May 1, 1983 for relief from the volumetric examination of the vessel to support skirt weld (Category B-H). Surface examination methods will be used in lieu of the volumetric methods required by the 1974 Edition Summer 1975 Addenda ASME Section XI. Relief will be requested because of the difficulty obtaining full volumetric coverage due to the geometry of the weld. New editions of the ASME code recognize this and permit surface examination.

II. Relief Requests From the September 10, 1979 ISI Program

Relief is requested from the requirement that all bolts, studs and nuts be inspected each inspection interval.

(1a) In lieu of this, bolted components which are disassembled for other maintenance or repair will have their bolts inspected.

This relief would apply to the Incore Monitoring Housing Assemblies and the Control Rod Drive (CRD) assemblies. To date, approximately 7 of 42 (approximately 16%) In-core Monitor assemblies (8 bolts per assembly) have been disassembled for maintenance and inspected. Approximately 79 of 137 (58%) Control Rod Drive Assemblies have been disassembled and inspected.

1b) Disassembly of the components discussed in item 1a above, solely for the inspection of bolting, is unjustified due to the following:

i) Loosening (and subsequent tightening) of bolting is the most common cause of bolting failure. The disassembly of these components for inspection will result in increased failure and additional radiation exposure. The disassembly area radiation dose rate is approximately 150-200 millirem/hour whole body.

ii) The increase in safety resulting from one hundred percent inspection is negligible. As stated in the James A. FitzPatrick Final Safety Analysis Report, Section 3.5 (pgs. 3.5-17, 35-18) each Control Rod Drive (CRD) housing bolt has an allowable load capacity of 15,200 lb. At reactor design pressure (1250 psig), the total load on all 8 bolts is 30,400 lb. Therefore, six of the eight bolts would have to fail to degrade the reactor coolant system boundary. Complete failure of all eight CRD housing bolts result in a maximum leakage of 840 gpm which is below the make-up capacity of the high pressure coolant injection pump. Such failure would not prevent complete and safe shutdown of the reactor.

Any leakage due to a complete or partial failure of these bolts would be detected by the Drywell Leakage Detection System.

The Incore Neutron Flux Monitor assembly has a diameter of 2" and the bolts are tack welded to prevent loosening. In the unlikely event of a bolt failure, a significant degradation of the coolant pressure boundary would not occur. Leakage from a complete failure of the in-core monitoring bolting, (assuming complete separation) would result in leakage less than the normal make-up rate provided by the HPCI systems.

1c) Table 1 of this Attachment is a list of reactor vessel pressure retaining bolts smaller than two inches in diameter.

2a) Figure 3.8-1 of the recently revised FitzPatrick FSAR is a sketch which indicates the relative locations of both the housing to vessel "J" weld and the stub tube to housing weld. Note also that these component penetrations are below the core support plate in the vessel bottom head. The difficulty of inspecting components below the core support plate prevents an adequate examination of the housing-to-vessel "J" weld.

Any Leakage due to a complete or partial failure of these welds would be detected by the Drywell Leakage Detection System.

In addition, the area beneath the vessel bottom head will be inspected for leakage during the ten year hydrostatic test.

2b) As discussed in the FitzPatrick FSAR, (Section 3.5.6.1) a complete circumferential failure of attachment weld would result in the separation of the housing from the vessel. The Control Rod Drive and housing would be driven into the CRD Support Structure (FitzPatrick FSAR Figure 3.5-3). In the worst case, this would result in a 0.06 inch diametrical clearance between the housing and reactor vessel. Reactor water would leak at a rate of approximately 440 gpm. Normal make-up would be provided by the Reactor Core Isolation Cooling System (RCIC) at 400 gpm, and the CRD pumps at 60 gpm.

Any leakage due to a complete or partial failure of these bolts would be detected by the Drywell Leakage Detection System.

3) We anticipate that all Main Steam Relief Valve bolting will be inspected within each ten year interval provided current valve testing requirements do not change.

There are four flanges which are part of the head spray portion of the RHR system. The same two flanges are disassembled during every refueling outage as part of the reactor disassembly. The bolting associated with these flanges will be inspected at least once per ten year interval. The bolting of the other flanges will be inspected only upon disassembly of these components for other maintenance or repair.

Two flanges serve as decontamination connections for each Recirculation System loop. Inspection of the "A" - loop four inch flange has already been performed. However, there is no scheduled maintenance for these components, bolt inspection will be performed only when disassembly is required

4a) Visual examination of the internal surfaces of the recirculation pumps is impractical because of the cost, difficulty in disassembly of the pump, and associated personnel radiation exposure. Disassembly, strictly for inspection has several detrimental effects:

i) A significant increase in radiation exposure to maintenance and inspection personnel. Radiation surveys performed at the exterior of the recirculation pump and the adjacent area indicate an average contact dose rate from 300-600 millirem/hr, with hot spots on both pumps from 1500 to approximately 2100 millirem/hr. These readings are for the exterior of the pump; readings on the inside of the casing (which is from 2 to 4 inches thick) would be significantly higher.

ii) Disassembly will increase the failure rate of internal pump components, which were designed to operate for forty years without replacement or repair of major components.

4b) The pump casing is a variable thickness casting. Ultrasonic wall thickness testing will be of limited value due to wall thickness variations. A casting of this thickness is unlikely to develop significant wall thinning or cracking. In addition, the high radiation levels at the pump exterior (300-600 millirem/hr.) would result in significant personnel radiation exposure during the lengthy disassembly, inspection and reassembly process.

4c) The reactor recirculation pump manufacturer (Byron-Jackson) confirmed in an August 4, 1982 telephone conversation the following:

i) Except for seals, the FitzPatrick reactor recirculation pump (DVSS 28" x 28" x 30") is designed to be essentially maintenance free.

ii) Byron-Jackson does not recommend disassembly soley for inspection. Their experience indicates that disassembly increases the frequency of component failure.

iii) Historically, inspections have revealed no degradation of interior surfaces.

iv) The only reported failure requiring pump disassembly was the cover to pump gasket.

4d) A limited nuclear utility industry survey revealed no utility who disassembles reactor recirculation pumps solely for inspection. It is unlikely that either FitzPatrick recirculation pump will have to be inspected during this interval.

5a) See response to item 1a above.

5b) Disassembly increases the bolt failure rate. Any leakage from the reactor recirculation system will be detected by the Drywell Leakage Detection System. Increased personnel radiation exposure will result from more frequent inspection. The Authority considers that disassembly, for bolt inspection alone, does not result in an increased level of safety.

The "A" recirculation pump mechanical seal has been replaced. The mechanical seal bolts were examined at that time. This represents an inspection of half the bolts within this category.

6a) See response to item 1a above.

6b) See response to items 1b and 5b above.

6c) A zero-degree ultrasonic test of a valve body would not be useful because:

i) It does not provide a reliable method for detecting shallow surface indications.

ii) It does not reliably detect material cracks, especially if these cracks are not nearly perpendicular to the sound beam.

iii) These valves are either cast or forged with a grain structure that causes severe ultrasonic sound beam attenuation. This will further reduce the ability to detect surface flaws.

iv) Many of these valves are made of stainless steel, which further attenuates the ultrasonic sound beam.

v) The Drywell Leakage Detection System provides a means for detecting valve leakage.

7) Under the postulated conditions of loss of coolant, from the three inch (I.D. of 2.9") reactor core isolation cooling (RCIC) steam line and from the three inch (I.D. of 2.63") Control Rod Drive return line, the reactor can be shutdown and cooled down in an orderly manner. In this event, make-up would be provided by the HPCI system using on site power as described in the James A. FitzPatrick FSAR Sections 6.3 and 6.4.

8) Additional time is required to research and evaluate the use of surface ultrasonic examination on 0.375 inch and 0.500 inch wall pipe. We are currently studying ultrasonic examination results for piping with wall thicknesses between 0.375 and 0.500 inch.

9a) Welds located outside and immediately adjacent to containment penetrations will be examined in accordance with code requirements. Either surface or volumetric examinations will be performed depending upon the category or class of the weld. These examinations will include the full length of the weld and will be performed at the required frequency.

9b) All containment penetrations are constructed to direct leakage into primary containment where it will be detected by the Drywell Leakage Detection S--tem.

The Drywell Leakage Detection System (DLDS) is described in the recently updated FitzPatrick FSAR, Section 4.10. Surveillance requirements and limiting conditions for operation of the DLDS are in the FitzPatrick Technical Specifications.

9c) A complete set of FitzPatrick Inservice Inspection Isometeric drawings were provided to Mr. George Freund of S.A.I. (P.O. Box 696, Idaho Falls, Idaho 83402) by letter dated August 20, 1982. These drawings illustrate containment penetrations and nearby welds.

9d) Listed below is the Code Category for welds where relief is requested:

System	Weld Code Category
Control Rod Drive	B-J
RCIC	B-J
HPCI	B-J
Main Steam	B-J
Feedwater	B-J
Core Spray	B-J
RWC	B-J
RHR	B-J and C-G [1]

[1] Refer to Inservice Inspection Isometeric drawings.

9e) There are no dissimilar metal welds included in this relief request.

10) Additional time is required to verify the geometry of these welds and to prepare sketches of their location.

Table 1

REACTOR VESSEL

PRESSURE RETAINING BOLTING SMALLER THAN 2 INCHES IN DIAMETER

In-Core Monitoring Penetrations
(8 bolts per assembly)

12-45	36-21
20-45	44-21
28-45	28-17
36-45	12-13
12-41	20-13
20-41	28-13
36-41	36-13
04-37	44-13
12-37	12-09
20-37	36-09
28-37	20-05
36-37	28-05
44-37	36-05
20-33	
28-33	
36-33	
04-29	
12-29	
20-29	
28-29	
36-29	
44-29	
12-25	
20-25	
28-25	
04-21	
12-21	
20-21	
28-21	

Control Rod Drive Penetrations
(8 bolts per assembly)

22-19	22-39	30-51	30-27
26-19	26-39	34-51	34-27
30-19	30-39	10-47	38-27
34-19	34-39	14-47	42-27
38-19	38-39	18-47	46-27
42-19	42-39	22-47	50-27
46-19	46-39	26-47	02-23
50-19	02-35	30-47	06-23
06-15	06-35	34-47	10-23
10-15	10-35	38-47	14-23
14-15	14-35	42-47	18-23
18-15	18-35	06-43	22-23
22-15	22-35	10-43	26-23
26-15	26-35	14-43	30-23
30-15	30-35	18-43	34-23
34-15	34-35	22-43	38-23
38-15	38-35	26-43	42-23
42-15	42-35	30-43	46-23
46-15	46-35	34-43	50-23
06-11	50-35	38-43	
10-11	02-31	42-43	
14-11	06-31	46-43	
18-11	10-31	06-39	
22-11	14-31	10-39	
26-11	18-31	14-39	
30-11	22-31	18-39	
34-11	26-31	02-19	
38-11	30-31	06-19	
42-11	38-31	10-19	
46-11	42-31	14-19	
10-07	46-31	18-19	
14-07	18-03	50-31	
18-07	22-03	02-27	
22-07	26-03	06-27	
26-07	30-03	10-27	
30-07	34-03	14-27	
34-07	18-51	18-27	
38-07	22-51	22-27	
42-07	26-51	26-27	