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New York Power Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

June 20, 1984
JPN-84-40

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
NRC Generic Letter #84-11
Inspection of BWR Stainless Steel Piping

References:
1. PASNY letter, J. P. Bayne to T. A. Ippolito, dated July 31, 1981 (JPN-81-54).
2. NYPA letter, J. P. Bayne to D. B. Vassallo, dated February 7, 1984 (JPN-84-08).
3. NYPA letter, J. P. Bayne to D. B. Vassallo, dated March 9, 1984 (JPN-84-16).
4. NYPA letter, J. P. Bayne to D. B. Vassallo, dated July 7, 1983 (JPN-83-64).
5. NYPA letter, J. P. Bayne to D. B. Vassallo, dated March 9, 1984 (JPN-84-16).

Dear Sir:

The Power Authority has been engaged in a program of inspection for, and mitigation of, intergranular stress corrosion cracking (IGSCC) in the stainless steel piping in the James A. FitzPatrick plant. The extent of this program has been described previously in Reference 1.

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Inspection of approximately 50% of the piping welds was performed during the 1983 refueling outage. These inspections were performed in accordance with the guidelines of I.E. Bulletin 83-02 and revealed no significant amount of IGSCC. Subsequently, induction heating stress improvement (IHSI) was successfully performed on eleven recirculation system welds in March, 1984. The results of the inspections performed during the summer of 1983 and March, 1984 have been transmitted via References 2 and 3. To date, there is only one confirmed instance of IGSCC, and one instance of an apparent indication of IGSCC which has been subsequently re-examined and evaluated as geometric reflection.

On the basis of inspections previously performed, and the successful application of IHSI to 11 welds in March, 1984, the Power Authority's plan and scope of future inspections are as follows (see Table 1 for detailed breakdown):

(a) Inspection of approximately fifty (50) recirculation system welds (or approximately 50% of the total welds in the recirculation system) in conjunction with the performance of IHSI on those welds scheduled for September, 1984.

(b) Inspection of the stainless steel piping described in Reference 1 in accordance with Generic Letter 84-11, paragraphs (a) - (d) during the refueling outage scheduled for February, 1985.

Although the specific variables that determine the occurrence of IGSCC in BWR piping have not been completely defined, there does appear to be a correlation with pipe diameter, geometry, weld preparation history and stress conditions. Indeed the original sampling plan of I.E. Bulletin 83-02 does differentiate inspections by pipe diameter and geometry. Therefore, the expansion of the inspection scope at the FitzPatrick facility will be in accordance with the intent of IWB-2340 subparagraphs (b) through (d) with the following modifications:

(1) Discovery of an IGSCC indication will require the inspection of an additional 20% of the welds in the same size piping as that in which the IGSCC was found.

(2) Discovery of a second indication of IGSCC by the additional examinations of (1) above will result in 100% inspection of piping of the same size as that in which the IGSCC indications were found.

The Power Authority considers that a 20% sample of each piping size, with emphasis given to those welds/joints most likely to experience IGSCC (due to high stress weld preparation, history of cracking at other plants, etc.) is sufficient to determine the existence of IGSCC in each particular size piping. Therefore, the expansion of the scope of inspections due to discovery of IGSCC will be restricted to the particular size piping in which an indication is found.

As recommended by paragraph 3 of Generic Letter 84-11, the level 2 and level 3 UT examiners involved in inspections for IGSCC will have demonstrated competence either in previous qualification under I.E. Bulletin 83-02 or via certification by successful completion of IGSCC detection and/or sizing courses at the EPRI NDE Center. At this time, there does not appear to be a critical shortage of qualified personnel, although this situation may change, due to outages at other plants, expanded inspection scopes, etc.

The Power Authority has previously submitted fracture mechanics analyses of apparent indications of IGSCC (References 4 and 5 which have been evaluated and accepted by the NRC staff. These submittals and any future analyses will address the concerns expressed in Attachment 2 of Generic Letter 84-11 regarding crack evaluation and repair criteria. Future remedial actions concerning the discovery of IGSCC will be considered on a case-by-case basis, depending on the extent and location of any cracking.

The Authority has developed and qualified weld overlay repair procedures and personnel and is actively involved in investigations of alternative repair methods. Additionally, replacement material and procedures have been or are in progress of being developed for certain limited runs of pipe, i.e., the core spray and reactor water cleanup piping. However, pending the outcome of the application of IHSI this September, the Authority is not planning to engage in complete or substantial replacement of the recirculation system piping.

Attachment 1 describes the JAF leakage detection system which is sufficiently sensitive to ensure detection and investigation of any unusual leakage. Additional leakage detection measures will be implemented as necessary, if the extent of IGSCC increases.

One of the two timers is set to alarm after the normal minimum amount of elapsed time between automatically initiated pump-out. If a pump-out is initiated before this timer signals, the operator is alerted to the possibility of high leakage rates.

A second timer is set to alarm after the normal maximum amount of time for an automatic pump-out at rated flow has elapsed. If the pump-out is not completed before this timer signals, abnormally high leakage rates or system maloperation (e.g., low flow rates) may be the cause.

Both pumps associated with a single sump discharge into a common discharge line. Associated with each of the two discharge lines is a flow element, signal conditioner, flow indicator, flow recorder and flow integrator which records the total integrated volume, in gallons, pumped from each sump. These flow integrators are currently used to establish the reactor coolant leakage rate inside the drywell in accordance with Technical Specification requirements.

In a letter dated September 28, 1981 (JPN-81-76), the Authority submitted proposed changes to the FitzPatrick Technical Specifications related to the implementation of NUREG-0313, Rev. 1. These proposed changes, and the other changes regarding inservice inspection, prescribe coolant leakage limits of 5 gpm for unidentified leakage, which is a 2gpm leakage increase within any 24 hour period and a total leakage not to exceed 25 gpm. The Authority is currently in the process of discussing these proposed changes with the NRC, and is preparing a new submittal that will supercede the 1981 changes.

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

Very truly yours,

J P Bayne

J. P. Bayne
Executive Vice President
Nuclear Generation

State of New York
County of Westchester

Subscribed and Sworn to before me this 19 day of June 1984.

Jeanne La Luna
Notary Public

JEANNE LA LUNA
NOTARY PUBLIC, STATE OF NEW YORK
NO. 60-4614305
QUALIFIED IN WESTCHESTER COUNTY
TERM EXPIRES MARCH 30th 19.85...

cc: Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, New York 13093

TABLE 1

Piping 1	Total Welds	Welds Insp. in Summer '83	Welds ** Insp. in 3/84	Additional Welds Insp. in 3/84	Welds Tentatively Scheduled for Sept., 1984
4" Recirc	18	5	0	0	0
12"	40	16	5	2	8
22"	16	12	1	0	15
28"	39	21	4	2	33
20" RHR	5	5	4	0	2
24" RHR	4	0	0	0	2, if possible
6" RWC	2	1	0	0	0
10" Core Spray	9*	5	0	0	0

* "A" side replaced; "B" side still considered susceptible

** Some of these welds were previously examined.

(1) This table does not include piping which has been removed or replaced.

ATTACHMENT 1

Leakage Monitoring
Primary Containment System

The reactor coolant leakage detection systems consist of the drywell sump monitoring system and the drywell continuous atmosphere monitoring system. The drywell continuous atmosphere monitoring system utilizes a three-channel monitor to provide information on particulate, iodine and noble gas activities in the drywell atmosphere. Two systems are provided to perform this function. This system supplements the drywell sump monitoring system to detect abnormal leakage from the reactor coolant system. In the event that the drywell continuous atmosphere monitoring system is temporarily inoperable, grab samples are taken periodically to monitor drywell activity.

The drywell (Primary Containment) is equipped with two sumps for the collection and removal of waste liquids. Each sump is serviced by two pumps.

The capacity of the drywell floor drain sump pumps is 100 gpm, and the capacity of the drywell equipment drain tank pumps is also 100 gpm.

The Drywell Equipment Drain Sump collects leakage from equipment such as controlled pump seal leak-off and valve stem leak-off lines that are piped directly to the sump. The floor Drain Sump collects leakage to the drywell floor or atmosphere from valve packing or other sources.

Both sumps are equipped with level sensors that are displayed in the Control Room. In addition to "high" and "high-high" annunciators, each sump level indicator continuously displays sump level and rate of sump level rise on a panel mounted dual trace strip chart recorder.

The drywell floor drain sump pump is started automatically when the liquid level in the sump initiates a high level switch. The pump is stopped automatically when the liquid drops to a point initiating the low level switch. The selected drywell equipment drain pump runs continuously recirculating the water in the sump through the equipment drain sump cooler. When a high level signal is received the sump discharge valve opens and the sump is pumped to the radwaste facility. The discharge valve closes on a sump low level signal.

Two interval timers per sump are used to detect off-normal leakage conditions by detecting either more-frequent-than-normal or longer-than-normal pump operation.