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July 16, 1979

U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region V
Suite 202, Walnut Creek Plaza
1990 North California Boulevard
Walnut Creek, California 94596

Attention: Mr. R. H. Engelken, Director

Dear Sir:

Docket No. 50-206
San Onofre Unit 1

By letter dated June 25, 1979, you forwarded IE Bulletin 79-13. The Bulletin requires action by licensees concerning the recent discovery of cracking in feedwater system piping at several nuclear power facilities.

Submitted herewith as Enclosure 1 is our response to IE Bulletin 79-13. The responses contained in Enclosure 1 correspond to the item numbers given in the Bulletin.

If you have any questions, or desire additional information concerning Enclosure 1, please contact me.

Sincerely,

J. H. Drake

Enclosure

cc: Director, Office of Inspection and Enforcement
Division of Reactor Operations Inspection

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AO/I

79-155

ENCLOSURE 1
RESPONSES TO IE BULLETIN 79-13 CONCERNING
FEEDWATER SYSTEM PIPE CRACKING
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

Items 1.a, b and c

Since all three steam generator feedwater nozzle to reducer and reducer to feedwater pipe welds were volumetrically examined during an outage of San Onofre Unit 1 in June of 1979, the inspection program required by these Items is not applicable to San Onofre Unit 1. The results of the examinations and the corrective action taken, or to be taken, as a result of the conditions found during the examinations performed during the June, 1979 outage are contained in the report entitled, "Interim Report, San Onofre Unit 1 Steam Generator Nozzle to Piping Inspection and Corrective Action, June 13, 1979". A copy of this report was submitted to the NRC Office of Inspection and Enforcement, Region V, by letter dated June 15, 1979 in Docket No. 50-206. As indicated in the June 15, 1979 letter, a final report containing the results of the complete metallurgical examinations will be submitted by July 30, 1979.

Items 2.a and c

Since the feedwater nozzles were volumetrically examined during the June, 1979 outage at San Onofre Unit 1, the inspection program required by these Items will be performed during the next refueling outage which is currently scheduled for the second quarter of 1980. As described in the report referenced in response to Items 1.a, b and c above, only the feedwater nozzle to reducer and reducer to pipe end welds were scheduled for volumetric examination during the next refueling outage. However, the inspection program required by these Items will be performed in lieu of the examinations described in that report.

Item 2.b

Since the San Onofre Unit 1 steam generators are not designed with separate nozzles for main feedwater and auxiliary feedwater, a response to this Item is not required.

Based on the categories of break sizes discussed above, the adequacy of station operating and emergency procedures to recognize and respond to a feedwater system pipe break has been evaluated. The results of the evaluation are summarized below:

1. <u>Large Breaks</u>

   The symptoms, confirmatory investigations, available instrumentation and operator actions utilized to recognize and respond to a large feedwater system pipe break are contained in Procedure S-3-5.20, "Steam Generator High Energy Pipe Break".

   The symptoms include decreasing steam generator level, steam flow versus feedwater flow partial matrix alarm, steam flow versus feedwater flow mismatch reactor trip, feedwater pump abnormal current, reactor coolant system pressure and temperature fluctuations, pressurizer level fluctuations, etc. The confirmatory investigations include examination of supplementary instrument indications to determine the cause/source of perturbations, etc. The available instrumentation is discussed in the response to Item 5.c, below. The operator actions include, reinitiation of normal feedwater or auxiliary feedwater, maintenance of reactor coolant system pressure, isolation of break, etc.

   This procedure will be revised to: (1) identify additional supplementary instrumentation for verifying large breaks, and, (2) provide a more simplified standard emergency response to large breaks.

2. <u>Medium Breaks</u>

   The symptoms, confirmatory investigations, available instrumentation and operator actions utilized to recognize and respond to a medium feedwater system pipe break are contained in Procedure S-3-5.7, "Abnormal Steam Generator Water Level".

   The symptoms include decreasing steam generator level, feedwater flow and steam flow fluctuations, feedwater pump abnormal current, decreasing condensate storage tank level, reactor coolant system pressure and temperature fluctuations, pressurizer level fluctuations, etc. The confirmatory investigations include examination of supplementary instrument indications to determine the cause/source of perturbations, etc. The available instrumentation is discussed in the response to Item 5.c, below. The operator actions include manual operation of feedwater regulating system, maintenance of steam generator water level, maintenance of reactor coolant system pressure, manual trip of the reactor, etc.

This procedure will be revised to identify additional supplementary instrumentation for verifying medium breaks.

3. Small Breaks

The symptoms, confirmatory investigations, available instrumentation and operator actions utilized to recognize and respond to small feedwater system pipe breaks are not explicitly contained in any procedure. However, the symptoms, confirmatory investigations, available instrumentation and operator actions utilized to recognize and respond to small reactor coolant system leakage inside containment as described in Procedure S-3-5.23, "Reactor Coolant System Leakage", would also result in recognizing and responding to small feedwater system pipe breaks inside containment.

The symptoms include abnormal sphere sump level and sump pump operation, sphere high humidity, sphere high pressure, etc. The confirmatory investigations include examination of supplementary instrument indications to determine the cause/source of perturbation, enter containment for inspection, etc. The available instrumentation is discussed in the response to Item 5.c, below. The operator actions include isolation of the break, bringing the reactor to a shutdown condition to perform corrective actions, etc.

Small feedwater system pipe breaks occurring outside containment would be detected by visual inspection in accordance with Technical Specification 4.10. The corrective actions to be taken in the event of detected leakage is also identified in Technical Specification 4.10.

Procedure S-3-5.23 will be revised to: (1) explicitly address small feedwater system pipe breaks, and, (2) identify additional supplementary instrumentation for verifying small breaks.

The procedures discussed above provide adequate instructions to recognize and respond to feedwater system pipe breaks. The procedural revisions discussed above are intended to increase the effectiveness of each procedure and are scheduled to be complete by September 15, 1979.

Item 5.c

The methods and sensitivity of detection of feedwater system pipe leaks (categorized as discussed in response to Item 5.b above) in containment are as follows:

Break Size	Method	Sensitivity
Large	Actuation of Reactor Trip Logic: o Steam Flow versus Feedwater Flow Partial Matrix Alarm o Steam Flow versus Feedwater Flow Mismatch Reactor Trip	Approximately 1000 gallons per minute or 500,000 lbs-mass per hour mass per hour or 25 percent of normal feedwater flow to each steam generator.
Medium	Recognizable Changes in Turbine Plant and Reactor Plant Systems Instrumentation: o Steam Generator Level o Steam Flow versus Feedwater Flow o Feedwater Pump Current o Condensate Storage Tank Level o Reactor Coolant System Pressure	Approximately 200 gallons per minute or 100,000 lbs-mass per hour.

Break Size	Method	Sensitivity
Small	Abnormal Indications of Process Instrumentation: o Sphere Sump Indication o Sphere Sump Pump Operation o Sphere High Humidity o Sphere High Pressure	Since small feedwater system pipe leaks would not produce recognizable changes in turbine plant or reactor plant systems, it is necessary to monitor abnormal instrument indications.* This method is currently used to detect small reactor coolant system leakage and is also used to detect small feedwater system pipe leaks. Abnormal instrument indication requires an investigation to determine the cause/source of the abnormality.

*The time required to detect small leakage inside containment is dependent on the size and location of the leak. For example, a 1 gallon per minute leak from the reactor coolant system would be detected by abnormal sphere sump pump operation within approximately six to eight hours.

Item 6

The results of the inspection program to be performed in accordance with the response to Items 2.a and c, above will be submitted within 30 days of completion of the inspection program to the Director, Office of Inspection and Enforcement, Region V, with a copy to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection.