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VPNPD-92-227 10 CFR 50.73
NRC-92-066

June 26, 1992

U. S. NUCLEAR REGULATOR COMMISSION
Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Gentlemen:

DOCKET 50-266
LICENSEE EVENT REPORT 92-005-00
EXCESSIVE COOL-DOWN TRANSIENT
POINT BEACH NUCLEAR PLANT, UNIT 1

Attached is Licensee Event Report 92-005-00, "Excessive Cool-Down Transient for Point Beach Nuclear Plant, Unit 1." On May 27, 1992, Refueling Procedure (RP) 6B, "Steam Generator Crevice Cleaning," was performed on Unit 1. The purpose of this procedure is to remove soluble materials from tubesheet crevice areas of the steam generators. During conduct of this procedure, the temperature cool-down limit defined in PBNP Technical Specification 15.3.1, "Reactor Coolant System," Specification B.1.b was exceeded.

An engineering analysis was performed to determine the potential effects of this temperature transient on the structural integrity of the limiting region of the reactor vessel. The results of the analysis concluded that the structural integrity of the reactor vessel is assured and that acceptable margins of safety will be maintained during subsequent operations.

This report is being submitted in accordance with the requirements in 10 CFR 50.73(a)(2)(i)(B), "The licensee shall report any operation or condition prohibited by the plant's Technical Specifications."



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If you need any further information, please contact us.

Sincerely,

Bob Link
Vice President
Nuclear Power

Attachment

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Point Beach Nuclear Plant, Unit 1 | 0 5 0 0 0 2 6 6 | 1 OF 0 5 |

TITLE (4): Excessive Cool-Down Transient

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAMES | DOCKET NUMBER(S) |
| 0 5 | 2 7 | 9 2 | 9 2 | 0 0 5 | 0 0 | 0 6 | 2 6 | 9 2 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): N

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

| | | | |
|---|---|---|---|
| 20.402(b) | 20.405(c) | 50.73(a)(2)(iv) | 73.71(b) |
| 20.405(a)(1)(i) | 50.36(c)(1) | 50.73(a)(2)(v) | 73.71(c) |
| 20.405(a)(1)(ii) | 50.36(c)(2) | 50.73(a)(2)(vii) | OTHER (Specify in Abstract below and in Text, NRC Form 366A) |
| 20.405(a)(1)(iii) | X 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| 20.405(a)(1)(iv) | 50.73(a)(2)(ii) | 50.73(a)(2)(viii)(B) | |
| 20.405(a)(1)(v) | 50.73(a)(2)(iii) | 50.73(a)(2)(x) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (AREA CODE) |
|---|---|
| J. C. Reisenbuechler, Manager - Oper. & Tech. Support | 4 1 4 7 5 5 - 2 3 2 1 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| X YES (If yes, complete EXPECTED SUBMISSION DATE) | NO | | 0 7 | 1 7 | 9 2 |

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

ABSTRACT

On May 27, 1992, during the performance of Refueling Procedure (RP) 6B, "Steam Generator Crevice Cleaning", the temperature cool-down limit defined in PBNP Technical Specification 15.3.1.B.1.b was exceeded. Technical Specification 15.3.1, "Reactor Coolant System," Specification B.1.b, defines a maximum cool-down limit of 100 degrees F in any one hour. The measured temperature cool-down between 0105 and 0205 was approximately 139 degrees F. An engineering analysis was performed to determine the potential effects of this temperature transient on the structural integrity of the limiting region of the reactor vessel. The results of the analysis concluded that the structural integrity of the reactor vessel is assured and that acceptable margins of safety will be maintained during subsequent operations.

EVENT DESCRIPTION

The purpose of Refueling Procedure (RP) 6B, "Steam Generator Crevice Cleaning", is to remove soluble materials from steam generator tubesheet crevice areas. Preparation for crevice flushing begins by filling the steam generators to approximately 80 inches, using the auxiliary feedwater pumps (AFPs) with the atmospheric steam dumps shut. The reactor coolant system (RCS) is heated to 325-335 degrees F and maintained at that temperature for at least two hours to dissolve contaminants in the steam generators. Both atmospheric steam dumps are then fully opened, and the steam generators allowed to boil for 30 minutes. Both atmospheric steam dumps are shut, and one RCP is secured. RHR is then used to perform a controlled cooldown to 175 - 190 degrees F in accordance with required cooldown limits. After obtaining chemistry samples, the steam generators are completely drained.

Following the preparation stage, the crevice flushing cycles are commenced. Using the AFPs, the steam generators are filled to 24-30" above the tubesheet. With both RCPs running and the atmospheric steam dumps shut, the RCS is heated to 290-300 degrees F and maintained at that temperature for 30 minutes. Both RCPs are then secured, and the RCS is lined up to one RHR pump with the heat exchanger bypassed to minimize cool-down. Both atmospheric steam dumps are fully opened, and the steam generators allowed to boil for 60 minutes. Cool-down of the RCS is a result of steam generator secondary side boiling and the addition of auxiliary feedwater necessary to maintain steam generator levels.

After boiling of steam generators for 60 minutes, the crevice flush cycle is completed by starting one RCP and cooling the RCS to 175-190 degrees F. Procedure RP-6B contains a number of notes which caution operators to the Point Beach Nuclear Plant administrative cool-down rate limit of 50 degrees F per hour.

The necessity to perform additional crevice flush cycles is determined by chemistry sample results. If additional crevice flush cycles are not required, Procedure RP-6B provides recovery instructions.

On May 27, 1992, crevice flush preparation stage had successfully been completed and Unit 1 was at 350 psig, 258 degrees F, with both RCPs running and both trains of RHR available. Heat-up of the RCS was in progress for the first cycle of crevice flushing. At approximately 0110, the 30-minute wait at 290-300 degrees F was completed and both RCPs were secured. As required by procedure, the steam generators were then boiled for 60 minutes, and at 0211 one RCP was started. It was later determined that a temperature transient occurred during this 60-minute steam generator boiling period. During performance of this crevice flush cycle, the operations crew was not aware that the

Technical Specification cool-down limit had been exceeded. The first cycle of the crevice flushing was completed at 0507 following the draining and refilling of the steam generators.

At 0921, during performance of the second crevice flush cycle of RP-6B, an operator noticed a higher than anticipated cool-down rate. The RCS had cooled down 60 degrees F in 25 minutes. The cool-down was stopped by securing auxiliary feed to the steam generators and starting a RCP.

At 1000, when reviewing cool-down data from the first crevice flush cycle, the control room Duty Shift Superintendent discovered that the RCS had been cooled down approximately 139 degrees F between 0105 and 0205, based on in-core thermocouple data retrieved from the plant process computer system. RCS pressure was 302-358 psig throughout the cool-down.

The event was reported to appropriate plant management, and the Technical Specification violation was acknowledged. Technical Specification 15.3.1.B.1.b limits the cool-down rate to less than 100 degrees F per hour. Although not required by 10 CFR 50.72, an informational notification was made to the NRC regarding this event at 1341 CDT on May 27, 1992.

CAUSE AND CORRECTIVE ACTION

A primary cause of this event is considered to be operator error. Control room operators failed to adequately monitor the cool-down rate during the crevice flush evolution.

Another contributing cause is an ineffective procedure. During the crevice flushing cycles, RP-6B requires both RCPs to be secured and the RCS lined up to one RHR pump with the heat exchanger bypassed. During the crevice flush evolution, leakage was occurring through the RHR heat exchanger outlet flow control valve. Therefore, during the crevice flush, three mechanisms were contributing to cool-down, including RHR heat exchanger cooling, boiling of the steam generators with both atmospheric steam dumps fully open, and filling of the steam generators with cool auxiliary feedwater. With no RCPs running and negligible decay heat, the procedurally required plant configuration may not have been appropriate to ensure that the cool-down rate could be effectively controlled or maintained within Technical Specification limits.

Procedural ineffectiveness, as well as other potential causes including concurrent work activities being performed by control operators, may also have contributed to the event.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
|---|---|---|---|---|---|
| Point Beach Nuclear Plant, Unit 1 | 0 5 0 0 0 2 6 6 | 1 | — 0 0 5 — | 0 0 | 0 4 OF 0 5 |

TEXT (If more space is required, use additional NRC Form 366A's) (17)

In response to this temperature transient event, a number of immediate actions were taken:

1. Steam generator crevice flushing activities were suspended. No additional crevice flushes were performed for the remainder of the Unit 1 outage.

2. An engineering analysis was initiated using ASME Section XI, Appendix E criteria to assess the potential effect of the temperature transient on the structural integrity of the reactor vessel beltline region.

3. A commitment was made not to pressurize Unit 1 above the Low Temperature Overpressure Protection limit until completion of the engineering analysis.

4. A review of data from the Fall 1991 Unit 2 Point Beach Refueling Outage concluded that no such temperature transient occurred during that outage.

5. An internal Incident Investigation Team was formed to further evaluate the causes of this event and to identify additional corrective actions.

6. The control board operator was removed from main control board watch-standing duties pending evaluation of his performance.

We are continuing our assessment of this event and attributable root causes which resulted in the temperature transient. Our Incident Investigation Team is expected to issue its evaluation report in early July. We expect to submit a supplemental Licensee Event Report in July 1992. This supplemental LER will further detail our assessment of the event and identify any additional corrective actions as appropriate.

SAFETY ASSESSMENT

ASME Section XI, Appendix E provides acceptance criteria and guidance for performing an engineering evaluation of the effects of an out-of-limit condition on the structural integrity of the reactor vessel beltline region. Adequate structural integrity of the reactor vessel beltline region is assured if it can be shown that the criterion specified in Appendix E is met throughout the event.

Our engineering analysis concluded that compliance with the ASME Section XI, Appendix E acceptance criteria was maintained. Therefore, our engineering analysis demonstrated that structural integrity of the reactor vessel beltline region is assured and that acceptable margins of safety will be maintained during subsequent operation.

NRC Form 366A (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES 8/31/85

# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
|---|---|---|---|---|---|
| Point Beach Nuclear Plant, Unit 1 | 0 5 0 0 0 2 6 6 | 9 2 | — 0 0 5 — | 0 0 | 0 5 OF 0 5 |

TEXT (If more space is required, use additional NRC Form 366A's) (17)

The health and safety of the public and plant personnel were not impacted by this event.

REPORTABILITY

This event is being reported in accordance with the requirements in 10 CFR 50.73(a)(2)(i)(B), "The licensee shall report any operation or condition prohibited by the plant's Technical Specifications."

GENERIC IMPLICATION

The concerns associated with excessive cool-down rates apply to all reactor pressure vessels.

SIMILAR OCCURRENCES

On May 10, 1991, while performing steam generator crevice flushing on PBNP Unit 1, the primary coolant temperature was not maintained in accordance with Technical Specification requirements. During this crevice flushing evolution, primary system temperatures exceeded 200 degrees F, as indicated by RHR heat exchanger inlet temperature, for approximately 17 minutes with a peak temperature of 201 degrees F. During this time, containment integrity was not set as required by Technical Specification 15.3.6.A.a. This event was reported in LER 91-004.