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SUBJECT:
LICENSEE EVENT REPT (RO 50-261/78-003) ON 01/29/78 CONCERNING PRIMARY SYSTEM PRESSURE EXCEEDED TECH SPEC LIMITS WHILE AT COLD SHUTDOWN PREPARING FOR INITIATION OF A MANUAL SAFETY INJECTION SIGNAL AS REQUIRED BY A REFUELING PERIODIC TEST... W/ATT SUPPORTIN

PLANT NAME: H B ROBINSON - UNIT 2

REVIEWER INITIAL: XJM
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February 13, 1978

FILE: NG-3516 (R) SERIAL: GD-78-404

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FEB 16, 1978
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. James P. O'Reilly, Director
U. S. Nuclear Regulatory Commission
Region II, Suite 1217
230 Peachtree Street, N.W.
Atlanta, Georgia 30303

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET 50-261
LICENSE NO. DPR-23
LICENSEE EVENT REPORT 78-003

Dear Mr. O'Reilly:

In accordance with Section 6.9.2.a of the Technical Specifications for the H. B. Robinson Steam Electric Plant, Unit 2, the attached Licensee Event Report is submitted. This report fulfills the requirement for a written report within fourteen (14) days of a reportable occurrence and is in accordance with the format set forth in Regulatory Guide 1.16, Revision 4.

Yours very truly,

B. J. Furr
Manager
Generation Department

CSB:as

Attachment

cc: Messrs. W. G. McDonald
E. Volgenau

7804180224

A002/S *
0/1

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

EXHIBIT A

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01	LICENSEE CODE (2)	S C H B R 2
	LICENSE NUMBER (3)	0 0 - 0 0 0 0 0 - 0 0
	LICENSE TYPE (4)	4 1 1 1 0
	CAT (5)	
CON'T 01	REPORT SOURCE (6)	L
	DOCKET NUMBER (7)	0 5 0 - 0 2 6 1
	EVENT DATE (8)	0 1 2 9 7 8
	REPORT DATE (9)	0 2 1 3 7 8

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 While at cold shutdown preparing for initiation of a manual safety
03 injection signal as required by a refueling Periodic Test, the primary
04 system pressure exceeded the limits of Technical Specification curves
05 of Figure 3.1-2. The primary system pressure peaked at 560 psig, 60 pounds
06 above the limit and the pressure dropped back below limits upon initiation
07 of the manual SI signal. No adverse impact resulted from this event.
08

09	SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP SUBCODE (15)	VALVE SUBCODE (16)
	C A	X	Z	V E S S E L	A	Z

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 8	—	0 0 3	/	0 1	T	—	0

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP SUPPLIER (25)	COMPONENT MANUFACTURER (26)
G	F	Z	Z	0 0 0 0	Y		N	C 4 9 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 The occurrence resulted from failure to adequately compensate for the
11 heat input from reactor coolant pumps and decay heat in the performance
12 of the refueling P.T. Procedural changes will be made and a mitigating
13 system will be installed during the current refueling outage to
14 effectively eliminate recurrence of such an event.

15	FACILITY STATUS (28)	D
	% POWER (29)	0 0 0
	OTHER STATUS (30)	N/A
	METHOD OF DISCOVERY (31)	A
	DISCOVERY DESCRIPTION (32)	Operator Observation
16	ACTIVITY RELEASED (33)	Z
	CONTENT OF RELEASE (34)	Z
	AMOUNT OF ACTIVITY (35)	N/A
	LOCATION OF RELEASE (36)	N/A
17	PERSONNEL EXPOSURES NUMBER (37)	0 0 0
	TYPE (38)	Z
	DESCRIPTION (39)	N/A
18	PERSONNEL INJURIES NUMBER (40)	0 0 0
	DESCRIPTION (41)	N/A
19	LOSS OF OR DAMAGE TO FACILITY TYPE (42)	Z
	DESCRIPTION (43)	N/A
20	PUBLICITY ISSUED (44)	N
	DESCRIPTION (45)	N/A
	NRC USE ONLY	

NAME OF PREPARER R. B. Starkey, Jr. PHONE (803) 332-1351

Supplemental Information
for
Reportable Occurrence 78-3

1. Report No.: 50-261/78-3

2a. Report Date: February 9, 1978

2b. Occurrence Date: January 29, 1978

3. Facility: H. B. Robinson Steam Electric Plant, Unit No. 2, Hartsville, South Carolina 29550

4. Identification of Occurrence: At approximately 1951 hours on January 29, 1978, the reactor coolant system pressure exceeded Technical Specification limits as defined by Limiting Condition for Operation 3.1.2.1 and associated Figure No. 3.1-2 for approximately 90 seconds. Peak pressure during this event was recorded to be 560 psig. This event is reported in accordance with Technical Specification 6.9.2.a.2.

5. Conditions Prior to Occurrence: The unit had been at zero power for approximately 44 hours. The unit was in cold shutdown with the reactor coolant system in solid condition. Pressure and temperature were being held constant at 360 psig and 155^{o}F respectively. Both "A" and "C" reactor coolant pumps were in operation for the purpose of circulating hydrogen peroxide (H_2O_2) solution. Prerequisites, initial conditions and valve lineups had been met and completed for the performance of the refueling Safety Injection Test (CPL-PT-2.1). Final preparations were being made prior to initiation of a manual safety injection.

6. Description of Occurrence: With the reactor coolant system in the condition as described above, the final steps prior to initiation of the manual safety injection were performed. The following describes the sequence of events surrounding and encompassing the occurrence.

At approximately 1950 hours, RHR discharge valve 744-B was closed and the running RHR pump "B" was shutdown. Immediately after stopping the RHR pumps, it was seen that a noticeable rise in reactor coolant system pressure was occurring. An attempt was immediately made to stabilize pressure rather than proceeding with manual safety injection. As soon as the pressure rise was recognized, the low pressure letdown control valve PCV-145 was manually opened to its full open position. In conjunction with this the RHR system to letdown line control valve RHR-HCV-142 was fully opened and the charging pump speed was reduced to the minimum level required to maintain Reactor Coolant Pump seal flow. After a brief observation period, it was determined that the pressure was not going to stabilize, and the pressure was approaching 500 psig (the Technical Specification limit). The decision was then made to immediately proceed with the initiation of safety injection in order to restart the RHR system. The manual safety injection was initiated at 1951. The RHR discharge valves started opening immediately, and both the "A" and the "B" RHR pumps were operating approximately 15 seconds later. During the time period the pressure exceeded 500 psig and peaked at 560 psig as recorded on the wide range PR-444 recorder. When the RHR pumps returned to service,

the pressure rise stopped and pressure dropped below the 500 psig limit. Total time above the Technical Specification limit was approximately 90 seconds. Attachment 1 contains graphical depiction of the occurrence as it appeared on the wide range pressure recorder.

7. Designation of Apparent Cause of Occurrence: The occurrence resulted from the pressure increase associated with the heat input to the system from the core decay heat and heat from the running reactor coolant pumps. The cause of the occurrence is attributed to the combination of insufficient instructions in the procedure and misjudgment on the part of the operators. The procedure performed (CPL-PT-2.1 Safety Injection Test) has as initial conditions that RCS pressure be at a level between 250 psig and 400 psig and the plant be in cold shutdown condition, on RHR, with RCP's running as required. These conditions were met when the test was performed as was described in part 6 of this report. After all initial conditions are met the final steps of the test in order are:

   1) Close RHR-744-B, RHR pumps discharge to safety injection system

   2) Stop the running RHR pump(s)

   3) Announce over P.A. system "Initiating Safety Injection", give countdown, then press either manual safety injection pushbutton thereby initiating a safety injection signal.

   All steps were followed in the order specified; however, the initiation of safety injection did not immediately follow stopping of the RHR pumps. With the concern for overpressurization within the industry and the overpressurization events at other utilities which had been thoroughly covered with the operators, it was considered prudent action when the pressure rise was realized to immediately take action to relieve or stabilize the pressure. The steps taken are described in part 6 above. When it was determined pressure would not stabilize, the decision was made to proceed with the initiation of the manual safety injection.

   The attempt to stabilize pressure coupled with the magnitude of the pressure rise and the initial conditions resulted in the pressure exceeding the Technical Specification limits.

8. Analysis of Occurrence: From the review and analysis of the occurrence it was determined that the pressure rise and resulting event was due to the input of decay heat and reactor coolant pump heat to the system. The percentages attributable to each were calculated to be approximately 77% decay heat and 23% reactor coolant pump heat. The analysis defining these percentages and the pressure rise associated with each of the probable causes of the pressure rise, i.e. charging flow (mass input), decay heat (heat input) and RCP (heat input) is attached. When comparing the total heat and mass input contribution toward

the total pressure rise, it is concluded that letdown isolation had not occurred and that a relief path did exist to reduce the magnitude of the overall pressure rise resulting from the heat input. However, it is apparent from the pressure rise that there was some restriction to the free release of the energy input. The pressure increase caused by this restriction to a free release of energy coupled with the time delay and initial conditions (pressure) of the test were sufficient to result in the transient exceeding the Technical Specification curve.

The pressure-temperature curves which appear in Technical Specification 3.1 indicate certain conservations which require review to determine if any structural limit, regarding the pressure vessel, was exceeded.

Figures 3.1-1 and 3.1-2 were generated by methods described in the bases of Technical Specification 3.1. In addition to the safety factors described in these bases, a 60 psi instrument error is included in the limits. Calibration of the Reactor Coolant System wide range and narrow range channels subsequent to the event indicate that the actual peak pressure achieved during the occurrence, based on the value recorded (560 psig) on the wide range recorder, was no greater than approximately 530 psig. The 30 psig difference was determined by applying a calibrated test signal to the recorder and comparing actual indication to the calibrated value. Additionally, adjustments were made to account for as-found data, during the transmitter calibration. An additional adjustment was made to account for instrument accuracy (2%) and the accuracy to which the recorder can be read (5 psig). In view of this calibration, the actual pressure experienced (530 psig) could be applied to the limiting value from the curve without the instrument error. This results in a maximum limiting pressure of 560 psig for the Technical Specification curve.

Another consideration regards the fact that the curves were generated to be valid for a vessel exposure equivalent to 20 effective full power years (EFPY) of operation. Based on this exposure, the curve maximum pressure for the temperature at the time of the occurrence is 500 psig. However, the current vessel exposure is approximately 5.0 EFPY. The curve which previously appeared as Figure 3.1-2, which was valid for an exposure equivalent to 4.25 EFPY of operations, resulted in a limiting pressure of 700 psig for the temperature in question. Interpolation for maximum pressure based on the actual exposure using the data above would result in a value well above the 530 psig addressed above. Although linear interpolation may not be totally valid in this application, it easily justifies the contention that no structural concerns exist.

Based on the above, although a Technical Specification limit was exceeded, no threat to the integrity of the pressure vessel occurred.

9. Corrective Action: Immediate corrective action involved the steps related in Section 6 of this report including initiation of the Safety Injection which terminated the pressure rise. Subsequent corrective action will include a thorough review of the event by each member of the operating staff. In addition the initial conditions of the subject periodic test will be revised to require that no reactor coolant pumps be operating. A precautionary note will be added to alert the operator that the steps which terminate cooling and initiate Safety Injection be performed as rapidly as system operation permits to avoid any excessive pressure rise due to the heat input to the system. Additionally, the entire procedure will again be reviewed to determine if any other changes may be made to reduce the potential for recurrence of such an event.

   In addition to the administrative and procedural changes, a Mitigating System, currently being installed, will provide sufficient capability for relieving a pressure rise of the magnitude encountered in this event. The system will provide a back-up to procedural and operator, checks and actions. It will provide alarming capability in addition to prompt and sufficient relief capacity to prevent pressure from exceeding the Appendix G limit. This modification will be completed during the current refueling outage.

   The above actions and modifications should effectively eliminate the potential for recurrence of this event in the future.

10. Failure Data: No occurrence of this type has previously been experienced at the H. B. Robinson Unit #2.

ATTACHMENT 1

TO

SUPPLEMENTAL INFORMATION

FOR

REPORTABLE OCCURRENCE 78-3

FEBRUARY 1978

ANALYSIS OF EVENT

J. M. Curley

February 1978

Analysis of Event

Summary

1. Sequence of events as recorded on figure 1.

2. Actual time - pressure rise observed and recorded on wide range recorder:

$$\frac{200 \text{ psi (increase)}}{90 \text{ sec. (period)}} = \underline{2.2 \text{ psi/sec.}}$$

3. Contributions to pressure-rise:

   a. Heat input: 3.1 psi/sec.
      77% attributable to decay heat
      23% attributable to pump heat

   b. Mass input:

      Single charging pump 1) high speed - 6.0 psi/sec.
      2) low speed - 2.4 psi/sec.

Conclusion

From the information obtained, it can be concluded that the event resulted from the heat input effects of both decay heat (approx. 77%) and reactor coolant pumps (approx. 23%). Since the heat and mass input effects would be additive in the event of letdown isolation, it is apparent that this isolation did not occur, i.e. the mass input was compensated by the letdown through 145 and 142. The analysis does indicate, however, that some restriction to the free release of energy due to heat input did occur. The pressure increase rate experienced is well within the requirements of the mitigating system which is presently in installation.

Analysis

1. Heat input - neglecting material mass

   A. RCP's (2)

$$2/3 \times 21.4 \times 10^6 \frac{\text{BTU}}{\text{hr.}} = \underline{\underline{1.43 \times 10^7 \text{ BTU/hr.}}}$$

   (Ref.: Westinghouse/CP&L calculated during plant startup tests)

   B. Decay heat 48 hours after S/D (Ref.: Bases Tech. Specs. Section 3.3)

   .62% of full power (2200 MWT)

$$.0062 \times 2200 \times 10^3 \times 56.92 \frac{\text{BTU}}{\text{MW-KW}} \times 60 = \underline{\underline{4.66 \times 10^7 \text{ BTU/hr.}}}$$

C. Total = 1 + 2 = $\underline{\underline{6.09 \times 10^7 \text{ BTU/hr.}}}$

D. RCS Volume:

$$\frac{9343 \text{ ft}^3}{.01632 \text{ ft}^3/\text{lbm}} = \underline{\underline{5.73 \times 10^5 \text{ lbm}}}$$

E. SP. Heat Input:

$$6.09 \times 10^7 \frac{\text{BTU}}{\text{hr.}} \times \frac{\text{hr.}}{3.600 \times 10^3 \text{ sec.}} \times \frac{1}{5.73 \times 10^5 \text{ lbm}} = \underline{\underline{\frac{.030 \text{ BTU}}{\text{lbm sec.}}}}$$

F. Sys. Temp. Increase Rate

$$.030 \frac{\text{BTU}}{\text{lbm-sec}} \times \frac{\text{lbm} - {}^\circ\text{F}}{.999 \text{ BTU}} = \underline{\underline{.030 \frac{{}^\circ\text{F}}{\text{sec.}}}}$$

G. Total Temp. Increase

Transient time = 90 secs.

$$90 \text{ sec.} \times .030 \frac{{}^\circ\text{F}}{\text{sec.}} = \underline{\underline{2.70 \ {}^\circ\text{F}}}$$

H. Corresponding Pressure Increase - Assuming Fixed Volume

From Steam Tables:
(360 psi)

$^\circ$F	v
160	.016381
152.7	.016343*
150	.016329

*Free expanding SP. volume - increase in v represents a pressure increase of 280 psi for fixed volume condition.

I. Pressure Increase Rate

$$\frac{280 \text{ psi}}{90 \text{ sec.}} = 3.1 \text{ psi/sec.}$$

2. Mass Input - 1 Charging Pump, No Letdown

A. RCS Mass: $\frac{9343 \text{ ft}^3}{.016329 \text{ ft}^3/\text{lbm}} = 572172.2 \text{ lbm}$

B. Mass Input Rates:

Case

1) Hi Speed $77 \text{ gpm} \times \frac{2.228 \times 10^{-3} \text{ ft}^3}{\text{sec-gpm}} \times \frac{\text{lbm}}{.0161 \text{ ft}^3} = 10.7 \frac{\text{lbm}}{\text{sec}}$

2) Lo Speed $30 \text{ gpm} \times \frac{2.228 \times 10^{-3} \text{ ft}^3}{\text{sec-gpm}} \times \frac{\text{lbm}}{.0161 \text{ ft}^3} = 4.15 \frac{\text{lbm}}{\text{sec}}$

C. Change in v - due to increased mass over 90 sec. period:

Hi Speed $\frac{9343 \text{ ft}^3}{572172.2 + (10.7)(90) \text{ lbm}} = .016302 \frac{\text{ft}^3}{\text{lbm}}$

Lo Speed $\frac{9343 \text{ ft}^3}{572172.2 + (4.15)(90) \text{ lbm}} = .016318 \frac{\text{ft}^3}{\text{lbm}}$

D. Pressure Increase:

Case: 1) Δ v from .016329 → .016302 540 psi
(360) → (900 psi)
2) Δ v from .016329 → .016318 220 psi
(360) → (580)

3. Comparison to Westinghouse Analysis:

A. From Westinghouse Mitigating System Analysis Results - July 1977

Mass Input - From Figure 3.1.2*

1) 16 lbm/sec*: $\frac{625-65}{40} = 14$ psi/sec.

2) 6 lbm/sec*: $\frac{620-65}{109} = 5.1$ psi/sec.

3) 113 lbm/sec*: $\frac{770-65}{6} = 117.5$ psi/sec.

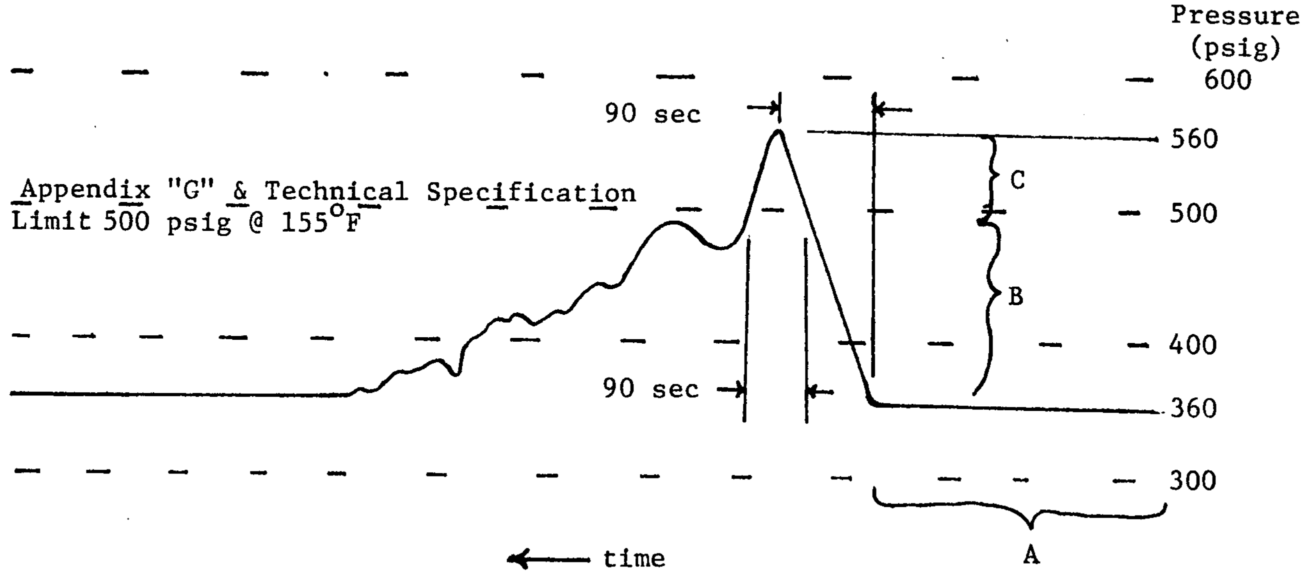
B. HBR Chg. Pump Speed Dependent Flow in

1) Hi Speed $77 \text{ gpm} \times \frac{2.228 \times 10^{-3} \text{ft}}{\text{sec-gpm}} \times \frac{\text{lbm}}{.0161 \text{ ft}^3} = 10.7 \frac{\text{lbm}}{\text{sec.}}$

2) Lo Speed $30 \text{ gpm} \times \frac{2.228 \times 10^{-3} \text{ft}}{\text{sec-gpm}} \times \frac{\text{lbm}}{.0161 \text{ ft}^3} = 4.15 \frac{\text{lbm}}{\text{sec.}}$

FIGURE 1

SEQUENCE OF EVENTS*



Period A: Prerequisites finalized, initial conditions met. System at 360 psig and 155°F.

Period B: (65 secs) 1) Close RHR 744B
2) Stop RHR Pump "B" 15 secs
3) Noted Pressure Increase
4) Manually Opened PCV-145
5) Opened HCV-142
6) Reduced Chg. Pump Speed to Minimum (30 gpm)

Period C: (25 secs) 1) Proceeded with Test Procedure
2) Initiated SI (Manually)
3) RHR 744 A & B Opened (12 secs)
4) RHR Pumps A & B Started (15 seconds after initiation)
5) Pressure Decreases

---

*Reference: Wide Range Pressure Recorder (From PT-402). January 29, 1978 @ 1900 - 2000 hrs.

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