NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)
Potential Residual Heat Removal System Overpressure During Accidents

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 NAME	DOCKET NUMBER
									PBNP Unit 2	05000301
04	03	97	97	-- 018 --	00	05	05	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 000

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)		Specify in Abstract below
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Glenn D. Adams, Licensing Engineer	(414) 221-4691

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
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 3, 1997, with Unit 1 in cold shutdown and Unit 2 in a defueled condition, licensee engineers discovered a potential for a section of the Residual Heat Removal (RHR) System inside containment to overpressurize during a design basis accident. The piping section is isolated by normally-closed RHR inlet isolation valves (RH-700 and RH-701), and is normally water-filled, but is not provided with relief valve protection. During a design basis accident which elevates containment temperature, the trapped fluid would be heated by the containment accident environment and could pressurize the isolated section. If unmitigated, the overpressure condition could lead to pipe rupture or valve damage, which would affect the capability of the RHR System to achieve and maintain cold shutdown if required later in the accident. This condition is a latent characteristic of the original RHR System design and is generic to both nuclear units. Prior to the startup of a nuclear unit, appropriate overpressure protection will be provided to that unit.

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

Event Description:

At 0819 CST on April 3, 1997, with Unit 1 in cold shutdown and Unit 2 in a defueled condition, licensee engineers discovered a potential for a section of the Residual Heat Removal (RHR) System inside containment to overpressurize during a design basis accident. The piping section is isolated by normally-closed RHR inlet isolation valves (RH-700 and RH-701), and is normally water-filled, but is not provided with relief valve protection. During a design basis accident which elevates containment temperature, the trapped fluid would be heated by the containment accident environment and could pressurize the isolated section. If unmitigated, the overpressure condition could lead to pipe rupture or valve damage, which would affect the capability of the RHR System to achieve and maintain cold shutdown if required later in the accident sequence. This condition is generic to both nuclear units.

The potential condition was discovered during an evaluation of Condition Report CR 97-0683, which described a recent plant operation that led to an unexpected alarm from the Low Temperature Overpressure Protection (LTOP) circuit. The evaluation determined that a brief pressure surge occurred in the Reactor Coolant System (RCS) when RH-700 was opened for the initiation of RHR operation. From this event it became evident that the RHR valves RH-700 and RH-701 were capable of trapping fluid in the isolated section. The fluid in the isolated section could be pressurized from leakby from the RCS during power operation, or it could be pressurized when the containment ambient temperature increase during plant startup causes the isolated water to expand. This latter type of overpressurization was described in NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions". An evaluation determined that the trapped fluid could heat up and pressurize the pipe beyond code allowable values.

The only accidents that may require the RHR System to operate in decay heat removal mode are the Steam Generator Tube Rupture (SGTR) and the Main Steam Line Break (MSLB) accidents. Of these, only the MSLB accident (if the break is inside containment) could raise the piping system temperature above normal ambient temperature. Therefore, the safety significance of the MSLB (inside containment) accident was evaluated as the limiting condition, and is discussed below.

In addition to thermally-induced isolated overpressure of the isolated piping section, the post-accident effects of thermal expansion were also reviewed with respect to NRC Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves". The failure of these RHR valves to open due to pressure locking could also have affected the capability to initiate RHR for a MSLB inside containment. In response to GL 95-07, plant modification requests were prepared to remedy the potential for pressure locking and thermal binding. These modifications were originally scheduled for completion during the

next scheduled refueling outages for Unit 1 (U1R24) and Unit 2 (U2R23).

The IEEE Standard 803A-1983 component identifiers for this report are:

Relief Valve (RV)
Valve (v)

Component and System Description:

The following discussion is generic to either nuclear unit.

The RHR System is an open-loop cooling system that draws reactor coolant from the hot leg of one reactor coolant loop and, after removing heat, returns the coolant to the cold leg of the opposite reactor coolant loop. There is a single inlet line from the RCS and one return line, which indicates that there was no original design provision to meet the single failure criterion of PBNP General Design Criterion (GDC) # 41. This decay heat removal loop of the RHR System was not originally considered an accident mitigating function because PBNP was generally considered a hot shutdown plant with respect to safe shutdown following accidents.

During a normal plant shutdown, the RHR System removes core decay heat and sensible heat after the secondary system has reduced reactor coolant system (RCS) conditions to approximately 350°F and 425 psig. To initiate RHR operation, the inlet isolation valves RH-700 and RH-701 must be opened to provide a flowpath from the RCS to the RHR pumps and heat exchangers. RH-700 and RH-701 are located in series, inside containment. After passing through the RHR heat exchangers, the coolant is returned to the RCS through return isolation valve RH-720 and a check valve.

RH-700 and RH-701 are motor-operated gate valves. During power operation, RH-700 and RH-701 are closed and electrical power is removed. When the RHR System is secured during a plant startup, the fluid in the isolated section could cool from 350 degrees F due to ambient heat losses.

As described in PBNP FSAR Chapter 14, the RHR system is also required to operate during a Main Steam Line Break (MSLB) accident and a Steam Generator Tube Rupture (SGTR) accident. To limit the offsite dose of the limiting MSLB (break outside containment), the FSAR analysis takes credit for cessation of steam release and the initiation of RHR within six (6) hours of the postulated accident. Similarly, the SGTR analysis in the PBNP FSAR also takes credit for the initiation of RHR within six (6) hours of the postulated accident.

A rupture of the piping between RH-700 and RH-701 would make the RHR system inoperable for use. This condition was discovered during evaluation of CR 97-0683 where it is believed that pressure buildup between 1RH-700 and 1RH-701 was released when 1RH-700 was opened and caused a Low Temperature Overpressure Protection (LTOP) alarm actuation.

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Cause:

Original design did not provide overpressure protection for the isolated piping section between RH-700 and RH-701 to accommodate the thermally-induced overpressurization that may occur during a design basis accident. Therefore, the original design did not provide ample assurance that the RHR piping inside containment would be available for all design basis events.

Corrective Actions:

1. An evaluation was performed to determine the potential stresses on the isolated piping section and determine Unit 1 and Unit 2 operability at those stresses. The predicted stresses exceeded the code allowable values.

2. *Prior to the startup of Unit 2 from the current refueling outage (U2R22), the potential for thermally-induced overpressure in the isolated RHR piping section will be remedied.* A solution under development is a plant modification (MR 95-041) to add bonnet vents to valves 2RH-700 and 2RH-701. This modification would vent each bonnet to the upstream side of the valve. The bonnet vent on valve 2RH-700 would also provide an overpressure relief path from the isolated section of RHR to the RCS. This modification would require the reactor unit to be defueled with reduced coolant inventory. This modification was initiated pursuant to NRC Generic Letter 95-07.

3. *Prior to the startup of Unit 1 from the current shutdown, the potential for thermally-induced overpressure in the isolated RHR piping section will be remedied.* Alternatives are presently being studied and include the installation of a relief valve on the existing vent and drain connections in the isolated RHR section. Installation of the bonnet vents to 1RH-700 and 1RH-701 is not feasible during the current outage because defueling is not feasible.

4. *During Unit 1 refueling outage U1R24, a plant modification (MR 95-042) is being considered to add bonnet vents to valves 1RH-700 and 1RH-701. This modification may serve as an alternative means to provide the necessary overpressure protection for the isolated section of RHR piping.* This modification was initiated pursuant to GL 95-07.

5. *A review will be conducted to identify any other potentially isolated sections inside containment that may affect the operability of safety-related equipment that is important to accident mitigation.*

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Reportability:

A 4-hour prompt notification per 10 CFR 50.72(b)(2)(i) was reported to the NRC duty officer at 1210 CST on April 3, 1997. This licensee event report is being submitted in accordance with the requirements of 10 CFR 50.73 (a)(2)(v)(D), "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident."

Safety Assessment:

An evaluation has shown that an increase in containment ambient temperature from shutdown temperature to the peak accident temperature could cause the water in the isolated section of piping between RH-700 and RH-701 to expand and cause a pressure that results in pipe stresses exceeding code allowable values. Of the two events that may require RHR operation post-accident, only the MSLB event (inside containment) would cause this thermal overpressure condition. The offsite dose consequences of this event would be small since there is no fuel failure from the event and any radionuclides released to the secondary would be retained in the containment. The FSAR analysis of the MSLB accident considers the break outside containment to be most radiologically limiting. The initiation of RHR operation may have been delayed indefinitely by the rupture of the isolated section (or by the pressure-locking of the isolation valves). In that case, core heat could have been removed by continued operation of the intact steam generator.

If the thermally-induced overpressure caused by a MSLB-inside-containment led to a rupture of the isolated piping section, it would not have led to a loss of coolant accident, because, by definition, isolation valve RH-700 would have to have been shut. Inadvertent opening of RH-700 in this condition is precluded by the normal isolation of power from the MOV. Therefore, it is not credible to postulate that the potential overpressure condition could have led to a MSLB and LOCA; an event for which the plant was not designed.

There would be no effect of this condition on the steam generator tube rupture (SGTR) accident because the SGTR accident does not cause the containment temperature increase that drives the thermally-induced overpressure condition.

Similar Occurrences:

Latent design flaws in the original design that affected the capability of safety-related equipment were reported in the following LERs:

LER	Description
266/97-006-00	Potential Refueling Cavity Drain Failure Could Affect

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Accident Mitigation

266/97-002-00	Potential To Overpressurize Piping Between Containment Isolation Valves During A Design Basis Accident
266/97-001-00	Safety Injection Delay Times Exceed Design Basis Values
266/96-005-00	Potential Service Water Flashing in Containment Fan Coolers