NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 2	05000237	1 OF 5

TITLE (4)
Containment Penetrations Outside Design Basis Due to Analysis of Thermally Induced Post-Accident Over-Pressurization

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
									Dresden Unit 3	05000249
01	27	97	97	-- 003 --	00	02	25	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Ram Mahendranathan - Sr. Design Engineer Ext. 3752	(815) 942-2920

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		07	11	97

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

Between October of 1996 and January of 1997, systems that penetrate containment were reviewed to identify piping that is susceptible to the thermally induced pressurization issue identified in NRC Generic Letter 96-06. Based on the results of the review and subsequent operability determinations, on January 27, 1997 it was determined that some of these systems identified as susceptible to over-pressurization in post-accident conditions did not account for post-accident pressurization in their original design, and as a result were outside the design basis of the plant.

Operability Assessments were performed to document the basis for continued operability of these affected penetrations and systems/components. The systems were determined to be operable but degraded.

Permanent correction of the degraded condition will be accomplished by installing over pressure protection or by penetration specific analysis. Alternative solutions are being evaluated and action taken will be reported in a supplement to this LER.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommendation Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Containment Penetrations Outside Design Basis due to Analysis of Thermally Induced Post-Accident Over-pressurization.

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2(3) Event Date: 1/27/97 Event Time: 1810

Reactor Mode: 1(4) Mode Name: Run(Shutdown) Power Level: 100(0)

Reactor Coolant System Pressure: 990(0) psig

B. DESCRIPTION OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B) which requires the reporting of any event or condition that resulted in the nuclear power plant: In a condition that was outside the design basis of the plant.

NRC Generic Letter 96-06 was issued on September 30, 1996 and identified a thermally induced over pressurization issue that could occur under post-accident conditions. The concern was that isolated water filled piping sections that penetrate containment could become over-pressurized following an accident that released energy to the containment, and subsequently jeopardize the ability of accident mitigating systems to perform their safety functions or fail, resulting in a breach of containment integrity.

Over the next four months, a review was performed to identify the piping sections for which this pressurization could jeopardize the ability of accident mitigating systems to perform their safety functions or could lead to breach of containment or bypass leakage.

Of the systems reviewed, twelve Unit 2 penetrations and eleven Unit 3 penetrations were determined to be susceptible to the over pressurization conditions discussed in the Generic Letter. The penetration and system descriptions are as follows:

X-139B, X-144 (Unit 2), and X-139C (Unit 3) - Control Rod Drive [CD]
X-119 (Unit 2 and 3) - Demin. Water Piping [KC]
X-109B (Unit 2) and X-109A (Unit 3) - Isolation Condenser Return [BL]
X-117 (Unit 2 and 3) - Drywell Floor Drain [WK]
X-118 (Unit 2 and 3) - Drywell Equipment Drain [WK]
X-122 (Unit 2 and 3) - Reactor Recirc. Sample Line [KN]
X-113 (Unit 2 and 3) - Reactor Water Cleanup [CE]
X-111A and X-111B (Unit 2 and 3) - Shutdown Cooling [BO]
X-123 and X-124 (Unit 2 and 3) - Reactor Bldg. Closed Cooling [CC]

These penetrations have valves on each side of the containment wall which are or may be closed during normal plant operation. The fluid trapped between the valves could be potentially heated during a loss of coolant accident or a main steam line break inside containment.

Since the post accident thermally induced pressures could be significant, the resultant pressure should have been included in the design basis of the containment penetrations. The significance of post accident thermally induced pressurization of containment penetrations was not known at the time the original design basis was established, and therefore, these pressures were not accounted for in the original design.

As systems were identified that could experience the thermal over-pressurization concerns, operability determinations were performed. These operability determinations were performed between January 7, 1997 and January 24, 1997 and determined the systems to be operable but degraded. The operability determinations were reviewed and the systems involved determined to be outside the design bases. The NRC was notified via the EMS phone on January 27, 1997 at 1843.

One system required additional immediate corrective action to support the operability evaluation. The piping between the closed containment isolation valves was drained on Units 2 and 3 Control Rod Drive System [CD]. On two Unit 3 systems, it was realized that additional margin could be gained, and as a result, the isolated portion of the Unit 3 Demineralized Water System [KC] between the closed containment isolation valves was drained and additional thermal insulation was installed on piping inside the containment on Unit 3 Reactor Building Closed Cooling Water System [CC]. For all other systems, no additional immediate corrective action was required.

Final corrective actions will be determined as part of the response to Generic Letter 96-06. Dresden intends to either install physical changes that provide over-pressure protection or use penetration specific analytical considerations that clearly address and confirm over-pressurization is not a concern. Alternative solutions are being evaluated to ensure the correct resolution for each isolated penetration is identified.

Discussion of specific changes and resolution for each affected penetration will be submitted in a supplement to this LER.

No systems or components were inoperable at the start of this event which contributed to the event. In addition, no manual or automatic engineered safety feature (ESF) actuation occurred as a result of this event.

C. CAUSE OF EVENT:

The cause of the event was Design Deficiency (NRC Cause Code B) in that the original plant design did not account for the possibility of overpressurization of isolated portions of piping penetrating containment. The thermally induced pressurization is a new issue identified in NRC Generic Letter 96-06 and the significance of it was not considered at the time the original design basis of the Dresden Station was established.

D. SAFETY ANALYSIS:

Dresden Operability Assessments 96-70, 97-10 and 97-11 were completed January 24, 1997 in accordance with Generic Letter 91-18 to document the basis for continued operability of these affected penetrations and systems/components.

The basis for operability of these affected penetrations includes consideration of one or more of the following: leakage from one of the containment isolation valves (seat, packing, or body to bonnet flange), insulation of piping to delay the temperature increase, lifting of air operated valves due to the pressure increase, draining of the isolated portion, and plastic straining of the affected pipe to accommodate the pressure increase.

If an accident had occurred which resulted in containment temperature increase, and subsequent isolated piping pressurization, these relief paths were available which would have made it less likely that piping or containment failure would have occurred. Had containment failure occurred that resulted in a release path to the Reactor Building, secondary containment and Standby Gas Treatment would have been mitigating factors for the event. Since the failure of the system boundary or an isolation valve would not necessarily cause a breach of containment, and given the redundant containment systems in place, the impact on the health and safety of the public is not significant. As a result, the safety significance of the event is minimal.

E. CORRECTIVE ACTIONS:

1. An extensive review of isolated piping sections that affect containment integrity has been performed. (Complete)

2. Operability determinations were performed on piping affected by post-accident thermal over-pressurization. (Complete)

3. The piping between the closed containment isolation valves was drained on Unit 2 and Unit 3 Control Rod Drive System [CD] and Unit 3 Demineralized Water System [KC]. Additional thermal insulation was installed on piping inside the containment on Unit 3 Reactor Building Closed Cooling system [CC]. (Complete)

4. To provide permanent corrective action, either over-pressure protection will be installed or penetration specific analytical considerations that confirm over pressurization is not a concern will be generated. The recommended corrective actions will be reported in a supplement to this LER. (NTS #2371809700301

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Dresden Nuclear Power Station, Unit 2	05000237	97	-- 003 --	00	5 OF 5

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

5. The modification process will be evaluated to ensure that future modifications account for overpressurization concerns. (NTS #2371809700302)

6. To ensure that appropriate lines are drained following future outages, procedures will be evaluated and changes made as necessary. (NTS #2371809700303)

7. Evaluate LLRT program and make changes as required to ensure that if existing credited valve leakage is corrected it is factored into the acceptance analysis. (NTS #2371809700304)

F. PREVIOUS OCCURRENCES:

No similar previous occurrences of this nature were identified.

G. COMPONENT FAILURE DATA:

Not Applicable