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NP-33-97-002

Docket No. 50-346

License No. NPF-3

February 21, 1997

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Ladies and Gentlemen:

LER 97-002
Davis-Besse Nuclear Power Station, Unit No. 1
Date of Occurrence - January 22, 1997

Enclosed please find Licensee Event Report 97-002, which is being submitted to provide 30 days written notification of the subject occurrence. This LER is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B).

Very truly yours,

JH Lash

James H. Lash
Plant Manager
Davis-Besse Nuclear Power Station

GMW/dlc

Enclosure

cc: Mr. A. B. Beach
Regional Administrator
USNRC Region III

Mr. Stan Stasek
DB-1 NRC Sr. Resident Inspector

Utility Radiological Safety Board

IE22

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

LICENSEE EVENT REPORT (LER)

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

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Davis-Besse Unit Number 1	05000 - 346	1 OF 6

TITLE (4)
Potential Overpressurization of Containment Penetrations During Accident Conditions

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
01	22	97	97	-- 002 --	00	02	21	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in text, NRC Form 366A)
			20.405(a)(1)(iv)	X	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 (Include Area Code)
Gerald M. Wolf, Engineer - Licensing	(419) 321-8114

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 January 22, 1997, with the unit in Mode 1 at approximately 100 percent power, it was identified that the design of some containment penetrations may not have adequately considered all possible accident conditions. This was based on review and evaluation of information contained in Information Notice 96-049, "Thermally Induced Pressurization of Nuclear Power Facility Piping," and Generic Letter 96-006, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." It was determined that six containment penetrations could exceed interim allowable pipe stress values during design basis loss of coolant accident conditions. Analysis showed that piping stress due to overpressurization is not sufficient to cause piping failure; therefore, no system or containment integrity would be lost. This condition is considered outside the plant design basis and is being reported in accordance with 10CFR50.73(a)(2)(ii)(B). Plant modifications and procedure changes are being developed to prevent overpressurization of the penetration piping.

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

Description of Occurrence:

On January 22, 1997, with the unit in Mode 1 at approximately 100 percent power, a condition was identified that was potentially outside of the design basis. After review and evaluation of information contained in Information Notice 96-049, "Thermally Induced Pressurization of Nuclear Power Facility Piping," and Generic Letter 96-006, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," it was determined that six containment penetrations could exceed interim allowable pipe stress values during design basis loss of coolant accident (LOCA) conditions. The NRC was notified of this condition at 1542 hours on January 22, 1997, via the Emergency Notification System in accordance with 10CFR50.72(b)(1)(ii)(B). This condition is being reported in accordance with 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis of the plant.

Apparent Cause of Occurrence:

The penetrations at the Davis-Besse Nuclear Power Station (DBNPS) were designed under American Society of Mechanical Engineers (ASME) 1971 Section III, Class 2 Boiler and Pressure Vessel Code requirements. This code required consideration of fluid pressure caused by heating of fluid trapped between two valves. These design considerations were left to the discretion and experience of the design engineers at the time of the original plant design. Pressure relief between closed containment isolation valves under accident conditions may have been considered unnecessary for a variety of reasons. However, for the subject penetrations, no evidence could be found that shows potential overpressurization of this piping was considered during the original plant design.

Analysis of Occurrence:

Davis-Besse identified thirteen water-filled containment penetrations that were susceptible to overpressurization as a result of being isolated and subsequently subjected to heatup from a post-accident containment atmosphere. Thermal expansion of the water could potentially result in pressures that create pipe stresses beyond ASME design allowable values. These penetrations were evaluated to determine the resulting pressures in order to identify required actions to mitigate the pressure increase. Preliminary calculations show that four of these penetrations meet ASME code piping stress allowables under the postulated conditions, three penetrations meet the Davis-Besse interim allowable pipe stress values (and are therefore acceptable for short term operation under the criteria established by IE Bulletin 79-14), and six penetrations exceed interim allowable pipe stress values.

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Analysis of Occurrence (continued):

In order to evaluate operability of the six penetrations exceeding interim allowable pipe stress values, the worst case overpressurization was considered. Specifically, worst case LOCA conditions were used, the piping between the isolation valves was considered water-solid, and it was assumed that there would be no leakage from the valves in the isolated piping to mitigate the pressure rise. For the most limiting penetrations the resulting pressures were determined to be sufficient to cause pipe stresses beyond yield. The volumetric expansion of the water could result in piping deformation, but not piping failure. Specifically, piping strain was predicted on the order of approximately two percent, while strain on the order of twenty percent would be necessary to cause piping failure. This led to a determination that no system or containment integrity would be lost, even without taking actions to mitigate the pressure increase. These penetrations are required only for containment integrity, and are not required for post-accident mitigation.

One of the penetrations (penetration number 74C) that meets interim allowable pipe stress values is the pressurizer auxiliary spray line (CB), which is credited for long term boron dilution in certain LOCA scenarios. Because of this function, the isolation valves must remain operable, even after a LOCA, when the resulting pressure could challenge valve operability due to pressure binding. To ensure the normally closed valves remain operable, the penetration piping was partially drained. This eliminates any potential for significant pressure increases and challenge to the operability of this flowpath. Even before this corrective action was taken, this penetration was considered operable. One of the two isolation valves remained capable of opening against this increased pressure, and it would have moved off its seat, relieving pressure and allowing the other valve to open before its operator would have been damaged by being stalled. For the other two penetrations that meet interim allowable pipe stress values (penetration 13, Containment Normal Sump (WK) and penetration 47A, Core Flood Sampling), no actions are required until the next refueling outage, when all three penetrations will either be modified, as necessary, to meet the ASME Code requirements, or will be protected from overpressure concerns by procedural changes.

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

Analysis of Occurrence (continued):

The preliminary evaluations have determined that the remaining six penetrations could exceed interim allowable pipe stress values. None of these penetrations support post-accident mitigating functions. The Component Cooling Water (CC) Outlet from containment penetration (4) has pre-existing, acceptable, quantified leakage as determined from local leak rate tests (LLRT) performed in accordance with 10CFR50, Appendix J. This leakage is expected to prevent overpressurization of this penetration. The Demineralized Water (KC) Supply to Containment penetration (21) and the Pressurizer Quench Tank Outlet (CA) penetration (48) have air-operated globe valves which are oriented such that pressurization of the penetration piping is projected to be relieved by causing the disk to be lifted against the actuator spring force without causing valve or piping damage. The Reactor Coolant System drain to the Reactor Coolant Drain Tank (CA) penetration (32) isolation valves are air-operated diaphragm valves. The diaphragm/air actuator would be slightly displaced by the pressure increase, allowing adequate leakage to prevent overpressurization. However, since this penetration does not need to remain in service, the penetration was isolated and verified to be partially drained.

The remaining two penetrations, Component Cooling Water Supply to the Control Rod Drive Motors (CD) (12) and the Refueling Canal Fill (CF) (49), would likely develop packing or valve seat leaks to limit the increase in pressure. However, without crediting this relief mechanism, the thermal expansion of liquid in these penetrations is projected to result in no more than approximately two percent plastic deformation under worst case conditions, and therefore, the affected penetrations remain operable. A modification is being developed for penetration 12 to protect the penetration from overpressure concerns. This modification will be installed during the next plant shutdown of sufficient duration for implementation, provided the engineering design work is complete and the necessary materials are available. This modification will be installed no later than the next refueling outage. The piping for penetration 49 was partially drained between the two closed isolation valves to reduce the volume of liquid subject to heatup and pressurization.

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Corrective Actions:

The Reactor Coolant System Drain to the Reactor Coolant Drain Tank Penetration (32) was isolated and the piping was verified to be partially drained on January 17, 1997, to ensure the piping is protected from overpressure concerns. This penetration is isolated by air operated valves, which preliminary evaluations have shown to provide ample inherent pressure relief. No further action will be required for this penetration.

On January 23, 1997, approximately one half gallon of water was drained from the Pressurizer Auxiliary Spray Penetration (74C) piping to prevent pressure binding of the penetration isolation valve.

On February 1, 1997, approximately two gallons of water were drained from the Refueling Canal Fill Penetration (49) piping. Analysis showed that its as-left LLRT leakage would have partially mitigated the overpressurization. Since this piping is only used during refueling outages, it was partially drained to provide an extra measure of overpressurization protection.

A modification is being developed to protect the Component Cooling Water Supply to the Control Rod Drive Motors Penetration (12). This modification will protect the penetration from overpressurization concerns and will be installed during the next plant shutdown of sufficient duration for implementation, provided the engineering design work is complete and the necessary materials are available. This modification will be installed no later than the next refueling outage, 11RFO, which is currently scheduled for spring, 1998.

The three penetrations (13, 47A, and 74C) which meet interim allowable pipe stress values, as well as the previously described penetrations (4 and 49) that do not meet ASME Code requirements, will either be modified or will be protected from overpressure concerns by procedural changes. These actions will be completed during the next refueling outage.

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Failure Data:

There have been no LERs in the previous two years involving the containment or containment penetrations that are considered to be potentially outside the design basis of the DBNPS. There have been six LERs regarding design basis issues in the previous two years. These LERs are as follows:

LER 95-001, Potentially Non-Conservative LOCA Analysis due to Modeling Errors
LER 96-002, Potential Loss of Remote Shutdown Capability due to MOV Fire Induced Damage
LER 96-004, Inadequate Compensatory Actions for Thermo-Lag for Radiant Energy Shields
LER 96-006, Reactor Coolant Pump Motor 2-1 Oil Collection System 1.5 Inch Lip Not Installed
LER 96-007, Control Room Emergency Ventilation System Design Bases Calculation Error
LER 96-010, Control Room Emergency Ventilation System Not Realized as Inoperable When Radiation Monitors Were Inoperable.

NP-33-97-002-0 PCAQR 96-1199