NRC Form 366
(9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	1 OF 0 4

TITLE (4)
BROKEN PIPE RESTRAINT REVEALS ERROR IN PIPING ANALYSIS

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)
										N/A	0 5 0 0 0
1 0	2 8	8 5	8 5	-	0 2 4	- 0 1	1 2	1 9	8 5	N/A	0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
0 9 6		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)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		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
	AREA CODE
W. K. Bandhauer, Nuclear Safety Supervisor	9 0 4 7 9 5 - 6 4 8 6

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	
B	B I	S P T	J 1 1 1								

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 fifteen single-space typewritten lines) (16)

On October 28, 1985, Crystal River Unit 3 was operating at 96% reactor power when a cracked concrete support pedestal for the seawater discharge piping from the A and B Nuclear Services Closed Cycle Cooling Water System heat exchangers (SWHE 1A and SWHE 1B) was discovered. Investigation into the cause of the crack revealed an error in the computer piping analysis. Expansion joints in the Nuclear Services Seawater System piping had been modeled incorrectly.

While the piping reanalyses were being performed, an additional problem was discovered. A rigid seismic restraint used in the computer model for the Nuclear Services Seawater System was not included in construction documentation and, therefore, was never installed.

Piping analyses were rerun and preliminary results indicated that the damaged pedestal could not withstand the revised design loads (pressure, deadweight and seismic). Further refinement of the piping analysis showed that the original (undamaged) pedestals were also not capable of withstanding the revised loads.

Modifications to reinforce the pedestals and reduce the pressure related loading effects of the expansion joint were completed for the Heat Exchangers on November 3, 1985. Installation of the missing seismic restraint was completed on November 4, 1985.

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EVENT DESCRIPTION

On October 28, 1985, Crystal River Unit 3 (CR-3) was operating at 96% reactor power when a cracked concrete support pedestal (BI, SPT) for the seawater discharge piping from the A and B Nuclear Services Closed Cycle Cooling Water System heat exchangers (BI, HX) (SWHE 1A and SWHE 1B) was discovered by an operator doing routine plant surveillance.

Engineering personnel performed a field walkdown and examined the problem on October 29, 1985. Following the walkdown, design load information for the cracked pedestal was requested from the CR-3 architect/engineer (A/E). While waiting for this information, available design loads for a similar pedestal on the seawater inlet side of the heat exhanger were reviewed. The loads were minimal and preliminary evaluation showed that the cracked pedestal would still provide adequate restraint.

On October 31, 1985, a second field inspection of the cracked support pedestal was performed. Based on this inspection of expansion joints, pipe routing and the support scheme, concerns were raised that a potential discrepancy between the analyzed design load and the actual load on this restraint might exist.

After the second field inspection, the requested design loads for the cracked pedestal were received from the A/E. The loads were of minimal magnitude and the A/E was asked to verify the accuracy of the existing computer analysis to address the concerns raised during the field walkdown. This verification effort confirmed that the Nuclear Services Seawater System piping model was deficient because there was an error in modeling the system expansion joints. At that time, Engineering and the A/E discussed design and repair alternatives and decided to prefabricate reinforcing plates for the cracked Nuclear Services Seawater System pedestals. These were to be used in the event that new analyses showed the cracked pedestal incapable of restraining actual loads.

On November 1, 1985, engineering personnel performed hand calculations to model the effects of previously omitted system pressure effect on the expansion joints. This analysis contained some very conservative assumptions to facilitate hand calculation. The most notable assumption was applying all of the torsion load from the corrected pressure term to the pedestal; the results of the analysis indicated that none of the pedestals in the system were adequate to restrain the increased loads. The A/E was directed to perform a new computer analysis using the original computer model and corrected pressure loading to determine how the new load would be distributed through the piping system and reduce loading on the pedestals. As a contingency, installation of the prefabricated pedestal restraints and expansion joint tie rods began.

Later in the day the A/E provided preliminary results of the reanalysis which showed a decrease in load when compared to the hand analysis. However, it was agreed that the damaged pedestal could not adequately restrain the revised loads while it was judged that the undamaged pedestals would be adequate to restrain the revised design loads. Heat exchangers SWHE 1A and 1B were declared inoperable because of the cracked support and revised analysis. Action Statement 3.7.3.1 was entered at 1815 hours. The A/E was directed to refine the analysis and determine the effect of the revised loads on the undamaged support pedestals. Installation of the pedestal restraints and expansion joint tie rods for inoperable heat exchangers continued and was completed on November 2, 1985. The Action Statement was exited at 0645 hours.

The refined analysis was completed on November 2, 1985 and showed that the undamaged support pedestals for heat exchangers SWHE 1C and 1D were not capable of withstanding the design loads. These heat exchangers were declared inoperable on the basis of the refined analysis and Action Statement 3.7.3.1 was entered at 1415 hours. These pedestals were repaired by installation of reinforcement plates and expansion joint restraints like those used on heat exchangers SWHE 1A and 1B. Repairs were completed and the Action Statement was exited at 1305 hours on November 3, 1985.

On November 2, 1985, during reanalysis of the Nuclear Services Seawater System piping, the A/E informed FPC that they had discovered an additional problem. A rigid seismic restraint, used in the computer model, had not been included in construction documents and, therefore, was never installed in the plant. The computer analysis was rerun without this restraint and the system was found to be unable to restrain seismic loads. This finding was reported to the NRC via telephone at 1855 hours. Temporary restraints were installed and remained in place until a permanent restraint was installed on November 4, 1985.

CAUSE

The cause of the cracked concrete support pedestal is attributed to a piping design error. The computer model input used to analyze the system did not properly model the pipe expansion joints. This error allowed moments caused by pressure forces to exceed the design loads of the pedestal. The cracks were probably induced by hydrostatic testing conducted during the 1985 refueling outage. It should be noted that hydrostatic test pressure (110 psig) is significantly in excess of the normal system pressure (one pump operation - 35 psig) or emergency system operation (two pumps - approximately 70 psig).

A review by the A/E of the circumstances surrounding the piping design error resulted in the filing of a report on December 9, 1985, per the requirements of 10 CFR 21.

The missing seismic restraint was the result of an omission during transfer of the computer design output to restraint fabrication documents.

To investigate the cause of this design error and determine if there were generic implications, the A/E conducted a review of 10% of the original piping support analysis packages. This review concentrated on work done by the originator and verifier of the package with the omitted restraint. Drawings used during this review were the seismic isometric drawings which were confirmed to as-built status as part of the IE Bulletin 79-14 effort. This review revealed no other examples of missing restraints.

During the operating life of the plant, approximately 13 additional percent of the piping analysis have been redone as part of the modification process. This redesign process typically includes a review of original design basis assumptions which would detect an error of this type. To date, none has been discovered.

From this investigation, the A/E concluded the event to be an isolated human error. To assure this assessment is correct, FPC engineering and quality assurance personnel conducted a design control audit of the A/E's design process and, specifically, the investigations conducted as a result of the missing restraint. The scope covered compliance with ANSI N45.2.11 design requirements as well as other areas. The audit team issued no findings relative to the missing restraint.

Also, since the original design documents were confirmed to be in agreement with the seismic isometric drawings, the validity of the work done in response to IE Bullegin 79-14 was reaffirmed to be correct and complete.

SAFETY CONSIDERATIONS

The Nuclear Services Closed Cycle Cooling Water Heat Exchangers provide cooling for the following essential components or systems upon receipt of an engineered safeguards signal:

Spent Fuel Coolers
Spent Fuel Coolant Pumps Air Handling Units
Reactor Building Fan Assembly Cooling Coils
Ventilation Fan Motors Coolers
Makeup and Purification Pump 1B
Makeup and Purification Pump 1A and 1C (Pump 1B Out of Service)
Steam Generator and Pressurizer Sample Coolers
Control Complex Water Chillers
Emergency Feedwater Pump Lube Oil Coolers
Air Coolers - Nuclear Services Closed Cycle Cooling and Seawater Pumps

The piping analysis error could have led to a failure which would render both trains inoperable. The missing seismic restraint, combined with a seismic event, could have also led to a similar event. The loss of both trains of Nuclear Services Closed Cycle Cooling would be a condition outside the design basis of the plant.

The design of piping systems is very conservative and the loss of both trains of these heat exchangers is considered extremely remote, even when the design errors are accounted for. This is supported by the fact that this system has operated continuously since the plant became operational and, at times, the system has operated with two seawater pumps running (Emergency System Operation). There have been no catastrophic failures.

CORRECTIVE ACTIONS

Repairs have been made to reinforce all of the piping support pedestals for the Nuclear Services Closed Cycle Cooling Heat Exchangers. Tie rods were installed across the Nuclear Services Seawater System piping expansion joints to eliminate the moment on the pedestals and a new seismic restraint was fabricated and installed in the location that was omitted in the original design.

An investigation is in progress by the A/E to determine the design adequacy for all safety related rubber expansion joints.

SIMILAR PREVIOUS EVENTS

Cracked Nuclear Support Closed Cycle Cooling Support pedestals have been found on two previous occasions. One event was attributed to concrete grout covering the support expansion slip joint and cracking under normal system expansion. The other event was attributed to damage caused during maintenance activities.

Florida Power
CORPORATION

December 19, 1985
3F1285-10

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Licensee Event Report No. 85-024-01

Dear Sir:

Enclosed is Licensee Event Report (LER) No. 85-024-01 which is submitted in accordance with 10 CFR 50.73.

Should there be any questions, please contact this office.

Sincerely,

G. R. Westafer

G. R. Westafer
Manager, Nuclear Operations
Licensing and Fuel Management

SCP/feb

Enclosure

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