U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit One	05000 269	1 of 8

TITLE (4) Non-Isolable Weld Leak On Pressurizer Surge Line Drain Pipe Causes Shutdown

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(S)
										05000
01	27	98	98	- 02 -	00	02	26	98		05000

OPERATING MODE (9): N

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	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 (*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)	
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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		04	30	98

ABSTRACT (*Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines*) (16)

On January 26, 1998, Unit 1 Reactor Coolant System (RCS) was at approximately 500 degrees F and 2100 psig. Heatup and pressurization of the RCS to hot shutdown was in progress. At approximately 0000 hours on January 27, 1998, upon noting an increase in the leak rate into the Reactor Building Normal Sump (RBNS), the operators began investigating for the leak source. At 0047 hours, a leak was discovered on the Pressurizer Surge Line drain line. This was identified as a non-isolable leak in the primary system and the Excessive Leakage Abnormal Procedure and Technical Specification 3.1.6.3 were entered. The RCS leak rate was calculated to be approximately 1 gpm. Unit startup activities were suspended and a shutdown was initiated. A one hour non-emergency NRC notification was made at 0138 hours. The Unit was safely and uneventfully shut down to the cold shutdown condition. Corrective action included removing and replacing the piping containing the leak and performing metallurgical examinations on the failure and adjacent pipe. The metallurgical analysis is not finalized. However, the preliminary metallurgical report has identified the root cause of the weld failure as externally initiated stress corrosion cracking (SCC). The preliminary report has also identified evidence of mixed mode SCC and fatigue propagation after the crack was initiated.

Background

Technical Specification 3.1.6.3 states that "If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection."

The drain line from the Pressurizer Surge Line (PSL) is Duke Class AC, i.e. Class A, since it is not isolated from the Reactor Coolant System (RCS)[EIIS:AB], but designed and analyzed to the requirements of Code Class 3 piping since it is less than or equal to one inch in diameter. The drain line was constructed from one inch diameter Schedule 160 Type 316 stainless steel piping having a 0.250 inch wall, with butt-welded 90 degree elbows. All four elbows were stamped with SA182 316H. Type 316H stainless steel is a high carbon version of Type 316 stainless steel with 0.04 to 0.1 weight percent carbon.

EVENT DESCRIPTION

On January 26, 1998, Unit 1 Reactor Coolant System (RCS) was at 500 degrees F and 2100 psig. Heatup and pressurization of the RCS to hot shutdown conditions were in progress. The Reactor Building Normal Sump (RBNS) was pumped at 2244 hours and again at 0000 hours. Upon noting an increase in leak rate, the operators began investigating for a leak source to the RBNS. At 0047 hours on January 27, 1998, a leak was discovered between the RCS Pressurizer Surge Line (PSL) and valve 1RC-17 (see Attachment 1) which is a drain line isolation valve. This was identified as a non-isolable leak in the primary system and the Abnormal Procedure for Excessive Leakage and Technical Specification 3.1.6.3 were entered. The RCS leak rate was calculated to be approximately 1 gpm. Unit startup was suspended at 0100 hours and a cooldown to cold shutdown was initiated at 0125 hours. A one hour non-emergency NRC notification was made at 0138 hours. The shutdown was accomplished expediently and uneventfully. The unit was completely depressurized and the pressurizer drained to isolate the leak. The water from the leak was contained within the RBNS and processed through the normal liquid waste processing systems with no problems.

On January 27, 1998, a Duke Power Failure Investigation Process (FIP) Team was assembled to determine the root cause of the failure, propose corrective actions, and support the plant recovery effort as appropriate based on information obtained during the investigation.

The team acknowledged that a major contributor to the determination of the failure mode would be the metallurgical analysis of the failed material in the PSL Drain. As a first step, a process for preserving the failed material was required. This included consideration of other potentially affected welds or pipe that could provide additional evidence.

The RCS system had to be cooled and drained. This is normally accomplished using the PSL Drain. However, there was a concern that using the drain line to cool down the RCS could cause the crack to open during the cooldown process. An evaluation was performed which demonstrated that the PSL drain line piping would thermally expand more than the PSL would be expected to move upward due to stratification. The net effect of the two movements would tend to close the crack. Normal cooldown and drain down methods were recommended and followed with no problems occurring.

The FIP team reviewed the Oconee Inservice Inspection (ISI) program documentation which indicated that all of the butt welds on the 10 inch diameter PSL have been examined since the unit began operation. No indications have been identified. The program also documented that there have been a total of 141 ISI examinations of Class A and B piping in the vicinity of Reactor Coolant Pump 1A1, which is adjacent to the Pressurizer, since the initial operation of Unit 1. None of these ISI welds have exhibited any type of indications.

A sample population of 1 inch Nominal Pipe Size (NPS) and smaller butt welded piping was identified for additional Penetrant Test (PT) examinations. The sample population included all welds that met the following criteria:

- welds in the 1A cavity
- Class 1 welds
- butt welds
- welds installed prior to the fire in 1973
- welds not already included in the ISI program (NPS 1 inch and smaller)

A total of 50 welds were identified by this process. Of the 50 identified, 14 were selected for Non-destructive Examination (NDE). Of those 14, eight were located on the PSL drain piping. All PT examinations for evidence of surface indications were negative.

The crack was located adjacent to weld 11, which joined a Schedule 160 Type 316 stainless steel vertical pipe section to a Type 316H stainless steel 90 degree elbow. It was decided to cut (at welds) and remove the line from the weld at the outlet of the PSL nozzle through the first four elbows including all straight pipe runs between the elbows. The two welds that were cut to remove the pipe were first PT'ed to determine if any surface indications existed. No indications were identified in the welds. A straight portion of the piping was also removed and retained at Oconee for additional PT work. No indications were identified in the piping. The remainder of the removed piping was packaged and shipped to the Lynchburg Research Center (LRC) of McDermott Technology, Inc. for further metallurgical examinations.

Visual inspection revealed the crack was located on the elbow side of the weld, extending approximately 170 degrees around the drain line outer diameter circumference, with the crack center located on the inner radius side of the elbow. When the fracture surfaces were exposed, it was determined that the cracking initiated on the drain line outer diameter from two distinct locations.

Tests also included stereovisual inspections, metallographic examination, Vickers microhardness measurements, scanning electron microscopy (SEM) and energy dispersive spectroscopy (EDS).

Examination by SEM and metallography indicated the crack propagated in a transgranular fashion essentially independent of the elbow material microstructure. More crack branching and secondary cracking was observed near the initiation sites than at the crack tip. These preliminary findings suggested crack initiation occurred by transgranular stress corrosion cracking. Evidence of fatigue striations at certain points away from the crack origins suggested mixed-mode crack propagation occurred after the initial crack development. The metallurgical report has not been finalized but indications are that this failure was the result of a combination of SCC and vibration.

The piping was removed from service and replaced with all new piping.

The Unit was returned to service. During unit heat-up, vibration was observed on the drain line but stopped after the unit reached stable hot shutdown conditions. This condition was evaluated and a monitoring program was established that will identify excessive vibrations should they occur. It also gave insight into the apparent mixed mode nature of the failure. Additional corrective actions are being considered.

CONCLUSION

The Operators very promptly identified the leak and safely shut down the Unit with no noted problems.

The preliminary metallurgical report suggests a mixed failure mode involving SCC and vibration. The root cause of the weld failure was identified as externally initiated stress corrosion cracking (SCC). The preliminary report has also identified evidence of mixed mode SCC and fatigue propagation after the crack was initiated. Based on these preliminary results and the Failure Investigation Process team findings, Units 2 and 3 should not expect to experience the same mixed mode conditions that led to the Unit 1 failure. Furthermore, the ISI program documentation and the additional 1 inch examinations performed provide information fortifying the conclusion that the crack in the drain line was an anomaly. Therefore, there was no operability concern with other piping in the 1A cavity.

A search and review of historic records and databases for information on past failures relative to this event was conducted. None of the past failures reviewed involved small bore, heavy wall, butt welded piping in the primary coolant system or another high pressure piping system. There was an event (LER 269/269/97-11) in that a Unit shutdown was initiated due to a Once through Steam Generator leak where a weld leak had failed due to SCC. Therefore, this event is considered to be recurring. However, corrective actions from that event could not have prevented this event.

There were no personnel injuries or personnel exposures associated with this event.

CORRECTIVE ACTIONS

IMMEDIATE

1. The leak was properly identified and the Unit was safely shut down.

SUBSEQUENT

1. Engineering reviewed the use of the Pressurizer Surge Line drain line for the required shutdown and made appropriate recommendations.

2. A vibration monitoring program was initiated to identify excessive vibrations should they occur.

PLANNED

1. Evaluate the value of including 100% of the non-isolable welds in an inspection program. Consider putting these welds in the "special-augmented-elective" category.

2. Evaluate the program which should be controlling introduction of corrosives to the plant for effectiveness and correct deficiencies as required.

Planned corrective actions 1 and 2 are considered to be NRC Commitment items. They are the only commitment items contained in this LER.

SAFETY ANALYSIS

The leak was calculated to be of a magnitude of approximately 1 gpm, which is well within the normal make-up capacity of a single High Pressure Injection (HPI) pump[EIIS:BG]. FSAR Section 15.14.4.3, "Small Break LOCA", defines the minimum area for a small break LOCA to be 0.007 sq. ft. This corresponds to a circular opening approximately 1.13 inches in diameter. If the Pressurizer Surge Line (PSL) drain line had broken completely at the weld failure, it would have produced an opening approximately 0.5 inches in diameter, which would produce a leak rate that would be within the capability of one HPI pump. Therefore, by definition, this event was not a LOCA. All identified consequences were bounded by UFSAR analyses for a Small Break LOCA. No Engineered Safeguards actuations were necessary as a result of the leak. All of the HPI system was at Engineered Safeguard readiness; therefore, the leak is not considered significant from a nuclear safety aspect.

In summary, the leak which occurred did not pose any hazard to the public. The leakage was contained. It is concluded that the health and safety of the public were not affected by this event.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION(4-95)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES:4/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oconee Nuclear Station, Unit One	269	98	02	00	8 of 8

ATTACHMENT 1

Diagram of Pressurizer Surge Line Drain Pipe