Duke Power®
A Duke Energy Company

D.M. JAMIL
Vice President

Duke Power
Catawba Nuclear Station
4800 Concord Rd. / CN01VP
York, SC 29745-9635

803 831 4251
803 831 3221 fax

November 9, 2004

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 2
Docket No. 50-414
Licensee Event Report 414/04-001 Revision 0
Reactor Coolant System Pressure Boundary Leakage
Due to Small Cracks Found in Steam Generator
Channel Head Bowl Drain Line on 2C & 2D Steam
Generators

Attached please find Licensee Event Report 414/04-001 Revision 0, entitled "Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2C & 2D Steam Generators."

This Licensee Event Report does not contain any regulatory commitments. Questions regarding this Licensee Event Report should be directed to R. D. Hart at (803) 831-3622.

Sincerely,

Dhiaa Jamil

Attachment

IE22

xc:

W.D. Travers
U.S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E.F. Guthrie
Senior Resident Inspector (CNS)
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

S.E. Peters (addressee only)
NRC Project Manager (CNS)
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop 10-B3
11555 Rockville Pike
Rockville, MD 20852-2738

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Catawba Nuclear Station, Unit 2	050- 00414	1 OF 6

4. TITLE
Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2C & 2D Steam Generators

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME NA	DOCKET NUMBER
09	16	2004	2004	– 001	– 00	11	09	2004	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	6
10. POWER LEVEL	0%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)	X	50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
R. D. Hart, Regulatory Compliance	803-831-3622

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B6a	RCS	NZL	West.	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On September 16, 2004, with Catawba Unit 2 in MODE 6 entering a refueling outage, a visual examination was performed on the steam generator (SG) 2A, 2C & 2D lower head bowl drains. Reactor Coolant System (RCS) pressure boundary leakage was identified on the 2C and 2D SG bowl drains. The 2A SG lower head bowl drain did not have any pressure boundary leakage. The pressure boundary leakage path was suspected to be the nozzle coupling to vessel weld. This event was reported to the NRCOC at 1131 on September 17, 2004 pursuant to 10 CFR 50.72 (b)(ii)(A). The most probable cause of the SG Bowl Drain leak is primary water stress corrosion cracking (PWSCC). The 2C & 2D SG bowl drain leaks were subsequently repaired, examined and tested. The 2A SG bowl drain was also repaired similar to the 2C & 2D SG bowl drains as a preventive measure. The 2B SG bowl drain had been repaired in 2001 due to a similar leak (LER 414/01-002). A corrective action from the previous LER was the development of a program to address Alloy 600 issues which included the inspections that discovered the events in this LER. This issue is not applicable to Unit 1 because the SGs are a different design that does not have a similar drain line. The overall safety significance of this event was determined to be minimal and there was no actual impact on the health and safety of the public.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	05000414	2004	001	00	2 OF 6

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

BACKGROUND

Catawba Nuclear Station Unit 2 is a Westinghouse Pressurized Water Reactor [EIIS: RCT]. Unit 2 has four steam generators (SG) [EIIS: SG] connected to the reactor coolant system (RCS) [EIIS: AB]. The Unit 2 SGs are Westinghouse Model D5 SGs. The four SGs are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles [EIIS: NZL] located in the hemispherical bottom head of the SG. The bottom head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. However, there is a small semi-circular hole at the center bottom of this plate to allow draining the bowl through one common drain line. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side. The SGs are primarily carbon steel. The heat transfer tubes are Inconel-600, the primary side of the tube sheet is clad with Inconel, and the interior surfaces of the reactor coolant channel head and nozzles are clad with austenitic stainless steel.

No structures, systems, or components were out of service at the time of this event that contributed to the event. This event is not applicable to the Unit 1 SGs because they are a different design and do not have a drain line in the bottom channel head.

The leakage of reactor coolant through the 2C and 2D SG channel head bowl drains was so minimal that it was detectable only by the visual observation of a small quantity of boron crystals. However, Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.13(a) limits RCS Operational Leakage to "No pressure boundary LEAKAGE" while in MODES 1 through 4. Condition B of TS 3.4.13 requires that if pressure boundary leakage exists, the unit is to be in MODE 3 within 6 hours and to be in MODE 5 within 36 hours. Therefore, Unit 2 operated in a condition prohibited by TS.

Unit 2 was operating in MODE 6, "refueling" immediately prior to this event. This event is being reported to the NRC pursuant to 10 CFR 50.73(a)(ii)(A), "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded." This event is also being reported pursuant to 10 CFR 50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical

Specifications" and 10CFR50.36(c)(2)(i), limiting condition for operation of a nuclear reactor not met.

EVENT DESCRIPTION
(Dates and times are approximate)

On September 16, 2004, visual and surface examinations of the SG 2A, 2C and 2D lower head bowl drains were conducted as part of the Alloy 600 Inspection program during the current Unit 2 refueling outage. This program was part of the corrective actions associated with the 2B SG lower head bowl drain leak that was reported to the NRC in LER 414/01-002, November 12, 2001. The bowl drains for SGs 2A, 2C and 2D were previously subjected to a penetrant test on March 3, 2003 and no rejectable indications were identified. A visual examination was performed on the SG lower head bowl drains and the 2C and 2D SG bowl drains were rejected. Surface examinations were performed on 2A and 2D. SG 2A lower head bowl drain was liquid penetrant tested and satisfied ASME Code acceptance limits. SG 2D lower head bowl drain was liquid penetrant tested and was rejected. SG 2C lower head bowl drain was not liquid penetrant tested since the visual inspection indicated an obvious pressure boundary through wall leak. This issue was documented in the Catawba corrective action program for resolution. An evaluation of this condition on September 17, 2004 determined that this event was reportable to the NRCOC pursuant to 10 CFR 50.72 (b)(ii)(A) "any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded." This event was reported to the NRCOC at 1131 on September 17, 2004.

The SG bowl drains are located in the center of the lower channel head. The opening to the SG bowl is beneath the partition plate that separates the hot leg side of the channel head from the cold leg side. The opening to the drain hole was measured to be 0.51" in diameter. A small passage in the partition plate above the drain hole called the "mouse hole" connects the bowl drain to the hot and cold leg channel heads simultaneously. The bowl drain was constructed by hard roll expanding an Inconel-600 sleeve into a clearance hole through the generator shell. The sleeve was seal welded to the stainless steel bowl cladding at the inner surface of the bowl. The lower end of the sleeve was seal welded to a butter layer of Inconel 82 or 182. A 316 stainless steel half-coupling was welded below the sleeve termination to form the bowl drain nozzle on the SG outer shell. The coupling was welded to the butter layer using a partial penetration weld and Inconel 82 filler material. A gap was left between the lower end of the sleeve and top end of the coupling to compensate for thermal expansion during welding. A 4" length of 3/8" diameter pipe connected the drain line to the drain nozzle. The drain line increased to ½" diameter pipe at a coupling below the 4" section and terminated at two valves and a pipe cap approximately 2-1/2 feet away from the drain nozzle.

CAUSAL FACTORS

The most probable cause for the SG 2C & 2D bowl drain leak is primary water stress corrosion cracking (PWSCC). The Alloy 600 weld filler material in the as welded condition has been shown to have a history of susceptibility to this type of degradation in the presence of primary coolant at PWR operating temperature. Axial cracks dispersed through the weld around the circumference of the nozzle demonstrated the generic susceptibility of the bulk filler material. No indications of circumferential extent were identified to suggest the structural integrity of the nozzle was challenged. The initiation of cracks at a gap exposing the back of the partial penetration weld to primary coolant was characteristic of PWSCC.

The existence of the gap between the nozzle coupling and lower end of the drain sleeve is likely the cause of this occurrence of cracking. This scenario meets all the requirements needed for PWSCC to occur, and is known through industry operating experience to be susceptible to attack. The weld material exists in a highly stressed state. The weld material is exposed to primary coolant via a gap, and the temperature is somewhere near the SG 2C and 2D hot leg temperature of 617 °F. Under these conditions, PWSCC is likely to occur.

Other forms of degradation were ruled out by a lack of evidence or discovery of a scenario consistent with those forms of degradation. The orientation and location of the cracking was inconsistent with several degradation modes such as mechanical or thermal fatigue.

CORRECTIVE ACTIONS

1. The SG 2A, 2C, & 2D channel head bowl drain lines were deleted and the drain line connection to the SGs was repaired and plugged to eliminate the PWSCC concern associated with alloy 600

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

weld material. The repair consisted of removing the indications by machining and welding in a new plug with weld filler material known to be resistant to PWSCC (Alloy 52).

2. In order to eliminate the potential exposure of carbon steel base metal from primary chemistry water, the sealing of the divider plate hole (mouse hole) for the 2A, 2C & 2D SGs was completed. The mouse hole modification is necessary to eliminate this concern and potentially damaging long term oxidizing effects from leakage through the upper seal weld or cracks in the Inconel 600 drain tube. This modification also serves to minimize dose during refueling outages by eliminating the crud trap in the bottom of the SG bowl. This modification had been previously completed for the 2B SG during the spring, 2003 refueling outage.

3. An Engineering Support Document was developed to address Alloy 600 issues as a corrective action to the previous LER (414/01-002). This includes an inspection program based upon a ranking of the susceptibility of Alloy 600 components. The Unit 2 SG bowl drains were included in this program as components with a high susceptibility requiring inspections.

This report does not contain any commitments required for regulatory compliance.

SAFETY ANALYSIS

With the completed modifications on SG 2A, 2C, & 2D bowl drains, there are no current operability concerns. The 2B SG was repaired during previous outages. No other methods of Non-Destructive Examination could be performed on these welds because of the geometry/configuration.

With this leak present during MODES 1 - 4, Technical Specification 3.4.13.a would not have been satisfied. No pressure boundary leakage is acceptable. The degraded condition of the bowl drain weld does not represent a challenge to the nuclear safety of the unit. The residual stresses from the partial penetration J-groove weld caused axial-radial cracks. These cracks grew until they resulted in a leak that was visible when the insulation was removed from the SG. The leakage did not exceed Technical Specification limits for unidentified RCS inventory loss, no radiation alarms sounded, and the small amounts of boric acid crystal deposits that were observed had caused no observable corrosion to the SG vessel. From a nuclear safety perspective, it has been concluded that any leakage due to typical PWSCC is not a challenge to nuclear safety. Reasons in support of this conclusion are:

- Leak rates from these cracks were very low
- Cracks were axial in orientation thus minimizing any potential for a catastrophic failure of a nozzle.

For the previous LER, a calculation was performed to determine the probable consequences on the RCS if the cracks on the bowl drain lines would not have been detected. The conclusion reached was that due to the cracks being in the radial/axial orientations, there would have been no catastrophic failure of the bowl drain connection. Engineering has evaluated the failures identified in this LER and no observations or other issues related to these failures has been identified or discovered that would lead to a different conclusion from that reached in previous LER. Therefore, the overall safety significance of this event was determined to be minimal and there was no actual impact on the health and safety of the public.

ADDITIONAL INFORMATION

The 2B SG bowl drain experienced a similar leak in the fall of 2001 and was reported to the NRC as LER 414/01-002 on November 12, 2001. The event reported in this LER has the same root cause. Therefore, this event was determined to be recurring in nature. However, this event only affected the SG bowl drains for Unit 2 and those SG bowl drains have been repaired to preclude recurrence. The Unit 1 SGs are of a different design and are not susceptible to this event. Catawba has implemented an Alloy 600 inspection program to monitor other areas that have Alloy 600. Therefore, no further corrective actions are necessary. Energy Industry Identification System (EIIS) codes are identified in the text as [EIIS: XX]. This event is considered reportable to the Equipment Performance and Information Exchange (EPIX) program. This event did not include a Safety System Functional Failure nor involve a personnel error. There were no releases of radioactive materials, radiation exposures or personnel injuries associated with this event.

NRC CORRESPONDENCE REVIEW AND CONCURRENCE FORM

Applicable Site(s) ONS ☐ MNS ☐ CNS ☒

Submittal Title/Subject: **LER 414-04-001, Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2C & 2D Steam Generators**

Background Information: ______

Scheduled Submittal Date: 11/09/04 Mandatory Submittal Date? (Y/N) N

PORC Approval Date 11/04/04 NSRB Approval Date NA Commitment Review Y/N

Submittal Lead Name Randall D. Hart Phone 831-3622

Content Development (Attach additional information as needed)

Contributor Name/Site	Scope/extent of Contribution or Portion Contributed
Randall D. Hart	Originator
W. O. Callaway	Engineering review

Individual Review (Attach additional information as needed)

Reviewer Name/Site	Signature	Portions or Subjects Reviewed and Nature of Review
Kay Nicholson	Kay Nicholson	General Content / Comitment

Regulatory Review

Reviewer	Name	Signature	Nature of Review
Submittal Lead	RD Hart	RD Hart	Entire document
NRIA/ RGC Manager	See attached		
SA Manager (Optional)	Thomas Ray	Thomas D. Ray	Entire document

Legal Dept. Review (Optional)

Name	Signature	Nature of Review
NA		

This completed form must be presented with the original submittal for signature. The form may be filled in with reference to e-mailed information in place of actual signatures and required information so long as the e-mailed information is maintained with the copy of this form that is sent to Master file.

Catawba Nuclear Station
LER 414/04-001-00
PIP C-04-04663

ENCLOSURE 1

Signature Sheet

Prepared By: ____________________ Date: 11/4/04

Reviewed By: ____________________ Date: 11/8/04

____________________ Date: 11/8/04

____________________ Date: 11/8/04

____________________ Date: ____________

Approved By: ____________________ Date: 11/8/04

ENCLOSURES:

1. Safety Review Signature Sheet
2. References
3. Corrective Action Schedule
4. Cause Code Summary
5. Personnel Contacted

Catawba Nuclear Station
LER 414/04-001-00
PIP C-04-04663

ENCLOSURE 2

REFERENCES

1. LER 414/04-001-00
2. PIP C-04-04663
3. Catawba TS 3.4.13 and Bases
4. LER 414-01-002
5. UFSAR Section 5.4.2
6. DBD CNS-1553.NC-00-0001, Rev. 14, Reactor Coolant System (NC).

ENCLOSURE 3

CORRECTIVE ACTION SCHEDULE

Corrective Action	Person(s) Contacted	Person(s) Assigned	Due Date
See PIP C-04-04663			

Catawba Nuclear Station
LER 414/04-001-00
PIP C-04-04663

ENCLOSURE 4

CAUSE CODE ASSIGNMENT SHEET

CAUSE CODE:

See PIP C-04-04663

ENCLOSURE 5

PERSONNEL CONTACTED

Personnel Contacted:

1. Dave Ward
2. W. O. Callaway