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A Duke Energy Company

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November 12, 2001

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 414/01-002 Revision 0
Reactor Coolant System Pressure Boundary Leakage
Due to Small Cracks Found in Steam Generator
Channel Head Bowl Drain Line on 2B Steam Generator

Attached please find Licensee Event Report 414/01-002 Revision 0, entitled "Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2B Steam Generator."

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,

G. R. Peterson

G. R. Peterson

Attachment

IE22

xc:

L. A. Reyes
U. S. Nuclear Regulatory Commission
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NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information 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 Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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.

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

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: None	DOCKET NUMBER
09	19	2001	2001	– 002	– 00	11	12	2001	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	5
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 19, 2001, with Catawba Unit 2 in MODE 5 preparing to enter a refueling outage, a walk down of steam generator (SG) 2B lower head bowl drain indicated boron residue buildup on the half inch piping immediately below the SG. The origin of the residue appeared to be adjacent to the bowl drain nozzle at the partial penetration weld between the nozzle coupling and the outer channel head surface. The pressure boundary leakage path was suspected to be the nozzle coupling to vessel weld. This event was reported to the NRC at 2212 on September 19, 2001 as an eight-hour non-emergency phone call pursuant to 10 CFR 50.72 (b)(3)(ii)(A). The root cause of the SG 2B bowl drain leak is primary water stress corrosion cracking (PWSCC) of Alloy 600 material. The 2B SG bowl drain leak was repaired and tested satisfactorily. The remaining three SGs on Unit 2 were visually inspected and liquid penetrant tests were performed. No similar leaks were detected. Long term corrective actions include evaluating SG drain line enhancements to preclude leakage and development of a program to address Alloy 600 issues. This issue is not applicable to Unit 1 because the SGs are of a different design that does not have a similar drain line.

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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 (PWR) [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 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.

On September 19, 2001, a walk down of SG 2B lower head bowl drain indicated boron residue buildup on the half inch piping immediately below the SG. The residue was white (as opposed to discolored) and had run down the approximately 6 inch length of vertical pipe below the outside shell surface. The total volume of boron residue was approximately 1 cubic inch. The origin of the residue appeared to be adjacent to the bowl drain nozzle at the partial penetration weld between the nozzle coupling and the outer channel head surface. There was a small amount of moisture visible on the surface at one location around the circumference covering an approximate arc of 20 degrees at the shell/nozzle weld. There was also boric acid residue on the piping to nozzle weld. The pressure boundary leakage path was suspected to be the nozzle coupling to vessel weld. The 3 other similar SG bowl drains were also visually inspected and liquid penetrant tests were performed. No evidence of boric acid residue or leakage was identified. This issue was documented in the Catawba corrective action program for resolution. This event was reported to the NRCOC at 2212 on September 19, 2001 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."

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

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 of a different design and do not have a drain line in the bottom channel head.

Although the leakage of reactor coolant through the 2B SG channel head bowl drain was so minimal that it was detectable only by the observed accumulation of boric acid crystals on the drain line, 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 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 5, "Cold Shutdown" 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 10CFR50.73(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 19, 2001, a walk down of SG 2B lower head bowl drain indicated boron residue buildup on the drain piping immediately below the SG. The total volume of boron was approximately 1 cubic inch. The origin of the deposit appeared to be at the partial penetration weld between the nozzle coupling and lower head shell. Visual inspection of the remaining three SGs detected no evidence of leakage or boric acid residue.

The 2B SG bowl drain is located in the center of the lower channel head. The opening to the SG bowl is obscured 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 inches in diameter. A small passage in the partition plate above the drain hole connects the bowl drain to the hot and cold leg channel heads simultaneously. The bowl drain was constructed

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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 82/182. 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-inch length of 3/8-inch diameter pipe connected the drain line to the drain nozzle. The drain line increased to half-inch diameter pipe at a coupling below the 4-inch section and terminated at two valves and a pipe cap approximately 2-1/2 feet away from the drain nozzle.

As part of the preliminary investigation, Quality Assurance inspectors conducted dye penetrant testing of the SG 2B drain nozzle and weld area. The test identified indications (flaws) at two locations. One was a pit-like indication located at the toe of the vessel to coupling weld on the vessel side of the weld. The dye produced a rounded indication of approximately 3/8-inch diameter. The second indication was located approximately 120 degrees counterclockwise from the first. A review of the drain line configuration determined that the most viable choice for evaluating the flaws was to examine the 2B SG bowl drain nozzle using a remote camera and video probe supplemented by dye penetrant testing where applicable. Inspection and testing were utilized throughout the removal process for the coupling so that all potential evidence could be gathered. Detailed characterizations of the defects occurred through iterative grinding and dye penetrant testing.

CAUSAL FACTORS

The root cause for the SG 2B bowl drain leak is primary water stress corrosion cracking (PWSCC) of Alloy 600 material. 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

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identified to suggest the structural integrity of the nozzle was challenged. The initiation of cracks, at the gap left between the lower end of the sleeve and top end of the coupling that exposed the back of the partial penetration weld to primary coolant, was characteristic of PWSCC. This scenario meets all the requirements needed for PWSCC to occur, and is known through industry contacts 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 2B 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. Lack of corroborative evidence ruled out contamination by other aggressive chemical species.

CORRECTIVE ACTIONS

Immediate:

1. A root cause team was assembled to assess the event including its cause(s), necessary corrective actions, and past/future operational impacts.

2. Visual inspections were performed on the SG bowl drain lines for SGs 2A, 2C, and 2D. These inspections found no evidence of boric acid residue or leaks.

Subsequent:

1. Dye penetrant tests were performed on the SG bowl drain lines for SGs 2A, 2C, and 2D. These tests found no indications.

2. The SG 2B channel head bowl drain line crack was repaired to eliminate the PWSCC concern associated with Alloy 600 weld material. The repair consisted of removing the indications and welding in a new coupling with weld filler material (Inconel 52/152) known to be resistant to PWSCC. The coupling was then plugged, removing the functioning drain from SG 2B. The

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geometry of the repair was selected so that the existing stress analysis on the SG would remain valid.

Planned:

1. Evaluate performing similar modifications to SG 2A, 2C, and 2D Bowl Drains to alleviate the Alloy 600 weld material concern of PWSCC. The modifications required to do this may also consider other factors in design including eliminating the "crud trap" that is currently present and also eliminating the small passage in the partition plate above the drain hole.

2. Evaluate performing a similar repair on SG 2B to eliminate the small passage in the partition plate above the drain hole. This will eliminate any issues associated with PWSCC of the seal weld.

3. Develop an Engineering Support Document to address Alloy 600 issues. This should include an inspection program, which may be based upon a ranking of the susceptibility of Alloy 600 components.

The planned corrective actions are being addressed via the Catawba Corrective Action Program. There are no NRC commitments contained in this LER.

SAFETY ANALYSIS

With the completed repair on SG 2B, structural integrity was restored and there are no current operability concerns. Visual inspections and penetrant tests (PT) performed on the other Unit 2 SGs showed no indications. No other methods of Non-Destructive Examination could be performed on these welds because of the geometry/configuration. This event is not applicable to the Unit 1 SGs because they are of a different design and do not have a drain line in the bottom channel head.

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

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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 are low
- Cracks are axial in orientation thus minimizing any potential for a catastrophic failure of a nozzle.

An evaluation was performed to determine the probable consequences on the RCS if the cracks on the SG 2B bowl drain line would not have been detected. The evaluation determined that the axial loads and bending loads on the half-coupling connection would be quite small. This along with the cracks being in the radial/axial direction results in a conclusion that there would have been no catastrophic failure of the bowl drain connection. In conclusion, 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

No events within the last three years have occurred involving RCS pressure boundary leakage at Catawba. There were no other LERs over the past three years that reported past PWSCC of Alloy 600 components or leaks at Catawba. A review of the corrective action program database also did not yield any events that reported past PWSCC of alloy 600 components or leaks. Therefore, this event was determined to be non-recurring in nature.

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.