Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

January 14, 1991
LIC-91-0003L

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

Subject: Licensee Event Report 90-28 for the Fort Calhoun Station

Please find attached Licensee Event Report 90-28 dated January 14, 1991. This report is being submitted voluntarily due to potential NRC and industry interest.

Further inspections resulting from this event and other activities planned for the next refueling outage will be discussed at a meeting with NRC personnel to be scheduled later in 1991.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/djm

Attachment

c: R. D. Martin, NRC Regional Administrator
W. C. Walker, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector
INPO Records Center



IE22 /1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Fort Calhoun Station Unit No. 1 | 0 5 0 0 0 2 8 5 | 1 OF 1 0 |

TITLE (4)

Leakage Through Control Element Drive Mechanism Housing

| 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) |
| 1 2 | 1 4 | 9 0 | 9 0 | — 0 2 8 — | 0 0 | 0 1 | 1 4 | 9 1 | N/A | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 2

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) | X 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) | 50.73(a)(2)(ii) | 50.73(a)(2)(viii)(B) | Voluntary Report |
| 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) |
|---|---|
| J. M. Cate, Special Services Engineer | 4 0 2 5 3 3 - 6 8 3 4 |

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 |
|---|---|---|---|---|---|---|---|---|---|
| X | A B | P S X | C 4 9 0 | Y | | | | | |
| | | | | | | | | | |

| 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 December 14, 1990, an investigation of unknown Reactor Coolant System (RCS) leakage identified the source as installed spare Control Element Drive Mechanism (CEDM) housing number 9. Subsequent removal and inspection identified two axial cracks in an inside diameter weld overlay region approximately two feet from the bottom flange of the housing. Similar installed spare CEDM housing number 13 was also removed and inspected, revealing two similar cracks in the weld overlay region.

The cause of this event was lack of venting, which created conditions conducive to transgranular stress corrosion cracking (TGSCC) in the spare housings. This report is submitted voluntarily due to potential NRC and industry interest.

Blank flanges were installed in place of CEDM housings 9 and 13. A procedure change has been implemented to assure complete venting of two other similar housings. Other appropriate CEDM housings have been examined with no cracks found. An enhanced RCS leakage monitoring program has been implemented.

At Fort Calhoun Station Unit No. 1, the Control Element Drive Mechanism (CEDM) housings are primary pressure boundary components. They were designed and fabricated to the requirements of the 1965 Edition, including the Winter 1967 Addenda, of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels. Each CEDM housing is mounted on a nozzle flanged pipe that is welded to the reactor vessel closure head.

The reactor vessel head nozzle flange is made of SA-182 Grade 316 stainless steel. The CEDM housings are fabricated from SA-182 and SA-312 Grade 347 or Grade 348 stainless steel. Each CEDM housing is omega-seal welded to the nozzle flange and then bolted to the nozzle flange with eight (8) threaded studs. Each stud is torqued in place with a hex nut over a pair of spherical washers.

As originally constructed, there were a total of forty-one (41) CEDM housings attached to the reactor vessel head. These forty-one (41) housings were identical in design to each other but utilized in different ways. Thirty-seven (37) of these locations have always been considered "active" CEDM housings since they house Control Element Drive Mechanisms which attach to Control Element Assemblies (CEAs). The remaining four (4) CEDM housings were installed spares originally designed for future use. Two (2) of these spares, at location numbers 7 and 1_ on the reactor vessel head, are now being used to house the Heated Junction Thermocouple (HJTC) probes. The other two (2) spare CEDM housings, at location numbers 9 and 13, contained only internal natural circulation spoiler assemblies and were essentially "passive" since initial plant startup. These two spare housings served no safety function other than maintaining the integrity of the primary pressure boundary.

On October 21, 1990, Reactor Coolant System (RCS) unknown leakage was identified and quantified at 0.1 to 0.2 gpm during operation at 100 percent power. Between October 21, 1990 and December 14, 1990, this leakage increased to and stabilized at approximately 0.4 gpm. During this period, extensive walkdowns of various plant systems including Reactor Coolant, Chemical and Volume Control, Safety Injection, Containment Spray, Sampling, and Waste Disposal Systems were performed. The source of the leakage, however, was not identified. The leak rate was verified by hand calculations using tank curves and verifying that the amount of water added to the RCS equaled the leak rate. Several possible leakage collection points were eliminated and it was determined that the leak was most likely an uncollected reactor coolant leak in containment.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
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| Fort Calhoun Station Unit No. 1 | 0 5 0 0 0 2 8 5 | 9 0 | — 0 2 8 | — 0 0 | 0 3 OF 1 0 |

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

On December 14, 1990, the reactor was placed in hot standby mode to look for the RCS leak on or around the Reactor Vessel head. An investigation team looking for the source of the unknown RCS leakage had narrowed the possibility for the leakage path to the Reactor Vessel head area. The receipt of alarms from fire detection instruments in the reactor vessel head area due to borated spray was further confirmation of the leakage location. The inspection of the head revealed a leak coming from the spare CEDM number 9 housing. The reactor was then placed in cold shutdown mode to allow further investigation and corrective actions.

On December 19, 1990, spare CEDM housing number 9 was removed, and a visual inspection was performed by ABB-Combustion Engineering (ABB-CE) personnel. Axially oriented cracks were identified on the inside diameter of the pressure housing, one of which had penetrated through-wall. The cracking was localized in a weld overlay area of the housing which exists on all the CEDM housings to provide positive positioning of applicable housing internals. On December 20, 1990, a 2.5 foot section of the housing containing the cracks was cut out and sent to ABB-CE facilities for metallurgical analysis.

On December 20, 1990, ABB-CE personnel performed an external visual inspection of CEDM housing numbers 1 and 4 for possible steam impingement damage, as these housings were located in the area where steam was leaking from CEDM housing number 9. No damage was found. The inspection team also completed an external visual inspection of CEDM numbers 7, 11, 14, 15, 17, 32, 34, and 38 to determine if any cracking was apparent on these housings. No defects were found on any of the housings that were inspected. Further investigation revealed that no damage was present on any other systems, the head, seismic skirt, CEDM housing externals, or fasteners.

As a result of the cracking found on CEDM housing number 9, the decision was made to remove CEDM housing number 13 from the reactor vessel head for a detailed examination, since it had been subject to the same conditions as housing number 9. On December 21, 1990, CEDM housing number 13 was removed and ABB-CE personnel performed an on-site visual inspection of the housing. The visual inspection of the number 13 housing also revealed axially oriented cracks in the area of the weld overlay, similar to those found on number 9. A 2.5 foot section of the number 13 housing containing the cracks was then cut out and sent to ABB-CE facilities for analysis.

CEDM blind flange assemblies were designed and fabricated by ABB-CE to replace the numbers 9 and 13 housings that had been removed. The blind flange assemblies were installed by ABB-CE personnel on December 27, 1990. The modification involved a change to the sealing mechanism from the original omega-seal welded design to a metal O-ring design. The CEDM flanges on the Reactor Vessel head have existing O-ring grooves which were originally used for the installation of blind flanges and O-rings used in initial hydrostatic testing of the head by Combustion Engineering. ABB-CE and Omaha Public Power District (OPPD) determined the acceptability of this modification for the remainder of Cycle 13 since there is no difference in the probability of occurrence or the consequences of primary coolant leakage from an omega-seal compared to an O-ring. Furthermore, in both sealing arrangements, the integrity of the joint is maintained by eight studs, spherical washer pairs, and nuts.

Upon receipt at the ABB-CE facilities in Windsor, CT, the sections of CEDM housings 9 and 13 were re-examined to verify the locations of the through-wall crack and the indications on the inside diameter of the housings. Two crack-like indications were identified in each of the housings. The housings were then sectioned to perform a more detailed visual inspection of the inside diameter surfaces. Visual exams were performed both with the naked eye and with low power magnification using a stereo microscope.

When short sections of the housings containing the indications were removed and cut longitudinally, the outside diameters decreased by 0.020 inches on number 9 and 0.023 inches on number 13. The nominal outside diameter is 8.627 inches and the nominal inside diameter is 7.189 inches. This decrease in diameter is attributable to residual stresses in the housing resulting from the weld overlay. The corresponding stress associated with the decrease in diameter was calculated to be on the order of 10 ksi. The tensile hoop stress introduced by an operational pressure of 2100 psi would be an additional 10.4 ksi which results in a total tensile stress in the weld overlay area of greater than 20 ksi. Under stagnant oxygenated conditions, this tensile stress level would be sufficient to result in transgranular stress corrosion cracking in the SA 312 Type 348 stainless steel pressure housing material. When a similar longitudinal cut was made on a section of the housing that did not contain the weld overlay, the measured diametrical change was only 0.0015 inches. This diametrical change indicates that the residual stresses in the housing material alone are quite low.

The portions of the housings containing the reducer section and the two full penetration butt welds were also sectioned longitudinally to establish whether there were any additional indications in other areas of the housings. The reducer sections were examined visually and a dye penetrant examination was performed on the reducer section from CEDM housing number 9. There were no indications revealed by these examinations in any location other than those previously found in the weld overlay.

Based on these examinations, only two (2) axially oriented cracks were confirmed in each housing. Number 9 had one (1) through-wall crack and one (1) crack which was approximately 85 percent through-wall. The through-wall crack had a length of approximately 2-7/8 inches on the inside diameter and 3/4 inch on the outside diameter. The two (2) cracks in the number 13 housing were determined to be approximately 95 percent and 70 percent through-wall. All four (4) cracks had aspect ratios (length on the inside diameter surface to depth of penetration) in the range of 3.7 to 3.9.

Fractographic examination of the cracks revealed all were initiated from the inside diameter of the housings. The initiation sites were all near the upper edge of the weld overlay region. The cracks then propagated outward into the wall of the pressure housing, extending nearly symmetrically downward through the weld overlay region and upward into the base metal of the pressure housing. The cracks were found to be nominally axial, but some of the cracks and portions of cracks were skewed off axial by approximately 15 degrees. The fracture surfaces of two (2) of the cracks had a clearly defined "ring" pattern that indicates that crack initiation occurred between 1981 and 1984. These dates were obtained by counting the number of rings observed on photographs of the fracture surfaces and then correlating each ring with one cycle of cold shutdown (with RCS drained down) and heat-up.

Scanning Electron Microscopy (SEM) of the crack surfaces and metallographic analysis of cross sections of the cracks were performed to identify the mode of cracking. The evaluations found all the cracking to be transgranular stress corrosion cracking (TGSCC). No impurity elements were found on the fracture surfaces. The types of austenitic stainless steels from which the CEDM housings are fabricated are known to be susceptible to TGSCC when exposed to adverse environmental conditions in the presence of tensile stress in the material. As discussed previously, it was found that the weld overlay in the CEDM housing introduced a significant residual tensile stress in the material.

To create the corrosive environment necessary for TGSCC to occur in the CEDM housings, high oxygen levels and some halogens (e.g., chlorides) must be present. Very low concentrations of chlorides can produce TGSCC when the oxygen content is high enough. However, the water chemistry at Fort Calhoun Station has been consistently controlled within Technical Specification limits. It was determined that with the chloride level within Technical Specification limits of less than 0.15 ppm, the oxygen level required to cause TGSCC is about 4 to 8 ppm. It was also determined that the installed spare CEDM housings, numbers 9 and 13, were not routinely vented during startup procedures during the previous operating life of the plant. It could not be positively determined why venting of these housings was not included in plant operating procedures or instructions. The estimated oxygen level in the spare CEDM housings without venting was calculated to be between 300 and 1300 ppm, which provided the conditions conducive to TGSCC.

To summarize, the cracks in the spare CEDM housings resulted from prolonged unvented operation which created conditions conducive to TGSCC. This report is submitted voluntarily due to potential NRC and industry interest.

The other two (2) spare CEDM housings with the HJTC probes, numbers 7 and 11, have been manually vented during startup since the HJTC probes were installed in 1984. Discussions between OPPD and ABB-CE revealed, however, that the venting procedures employed may not have ensured that these housings were free of air bubbles. The procedures did not ensure that venting would take place after the starting of the reactor coolant pumps during heatup. It was postulated that, if the HJTC housings were vented prior to starting the reactor coolant pumps, air bubbles from the steam generator tubes could become trapped in the HJTC housings when the pumps were started. Based on this information, it was decided that housings 7 and 11 would be examined by ultrasonic testing (UT) to determine the presence of cracks. This UT examination, utilizing both shear wave and refracted L-wave techniques, was performed by EBASCO Services personnel on December 29, 30, and 31, 1990. No crack indications were found.

The remaining 37 active CEDM housings are self venting through the rotating mechanical seals in the CEDM seal housing. Also, when the CEDMs are operated, there is an interchange of coolant water between the housing and the bulk RCS coolant inventory. As a result of the venting and the interchange of coolant during CEDM operation, the oxygen levels in the active housings should closely reflect the low oxygen levels of the bulk RCS inventory. Therefore, TGSCC in active CEDM housings is not considered credible.

In addition to the forty-one (41) CEDM housings, there are six (6) In-Core Instrumentation (ICI) housings located on the reactor vessel closure head. These housings have also not been vented since initial startup in 1973. Despite this fact, the ICI housings are not considered to be susceptible to the same kind of stress corrosion cracking observed on the spare CEDM housings because the ICI housings do not have a weld overlay region. As a result, there are lower residual stresses to assist in the initiation of stress corrosion cracking. Additionally, the ICI housing diameter is approximately a factor of 2 larger than the CEDM penetration diameter. This promotes more naturally convective coolant circulation so that internal oxygen content is closer to that of the bulk RCS coolant inventory. Thus, the ICI housings are not considered to be subject to TGSCC.

OPPD determined that there was minimal safety significance associated with the cracks in the number 9 spare CEDM housing. This determination was based on the individual assessments noted below of (1) reactor coolant system leakage, (2) potential for catastrophic rupture, (3) steam impingement, and (4) boric acid corrosion .

(1) Reactor Coolant System Leakage

The spare CEDM housings (numbers 9 and 13) did not have a safety function other than maintaining the integrity of the primary pressure boundary. Since there is no means during operation of detecting leakage as being specifically from a CEDM housing, any leakage from a CEDM housing would be categorized as from an unknown source.

To assure safe reactor operation, the reactor coolant system leakage limit from an unidentified source is limited to 1 gpm by Technical Specification 2.1.4. If the unidentified leakage exceeds 1 gpm, the reactor must be in hot shutdown within 12 hours and cold shutdown within 24 hours. Reactor coolant leakage indicates the possibility of a breach in the primary pressure boundary. The basis for the low leakage limits is to minimize the chance of a crack progressing to an unsafe condition without detection and proper evaluation. When the source of the leakage is unknown, placing the reactor in hot shutdown within 12 hours provides adequate time for an orderly reduction of plant power level. The hot shutdown condition also allows personnel to enter the containment and inspect the pressure boundary for leaks. The 24 hours allowed prior to going to cold shutdown allows reasonable time to correct small deficiencies. If major repairs are needed, a cold shutdown condition would be in order.

During this event, the reactor was shut down and eventually placed in a cold shutdown condition with RCS leakage well below the Technical Specification limit. The axial orientation of the cracks on the housing resulted in a slowly increasing rate of primary coolant leakage which was monitored and also provided sufficient time to place the reactor in a safe shutdown condition. Thus, the consequences of the RCS leak were well within the licensed design basis of the plant.

(2) Potential for Catastrophic Rupture

The potential for a catastrophic rupture from the stress corrosion cracking was evaluated. The stress corrosion cracks were oriented axially along the housing. This crack growth orientation does not readily lend itself to sudden crack growth and rupture. Austenitic stainless steel is sufficiently ductile such that rapid crack propagation would not be likely before the reactor could be shut down in an orderly manner due to excess leakage.

(3) Steam Impingement

Steam sprayed from the through-wall crack in the number 9 CEDM housing onto adjacent active CEDM housings numbers 1 and 4, potentially causing impingement damage to these active CEDM housings. The active CEDM housings have mechanisms with CEAs that control the reactivity in the reactor during normal operation, postulated accidents or other potential malfunctions. These active CEDM housings thus include equipment that is important to safety.

The adjacent CEDM housings, including numbers 1 and 4, were visually inspected for indications of steam impingement damage. No impingement damage was detected.

(4) Boric Acid Corrosion

A potential problem for reactor equipment is corrosion wastage which can result from the leakage of borated primary coolant water. Evaporation of this water leaves dry crystalline boric acid residue which is essentially non-corrosive. However, any subsequent re-wetting of this residue creates a boric acid slurry that causes corrosion wastage.

The CEDMs adjacent to the through-wall crack and a few locations on the reactor vessel head were visually inspected for damage from the boric acid residue. No damage was detected. A large amount of boric acid residue was cleaned up from accessible areas during this inspection.

A review of the Radiation Work Permits written as a result of this event indicated that approximately 13.3 man-rem of whole body gamma radiation exposure and approximately 0.5 man-rem of assigned skin dose were received by personnel involved with inspection and repairs at the site. There were no internal doses recorded. Workers receiving doses included personnel from OPPD's Pressure Equipment, Electrical Maintenance, Mechanical Maintenance, Operations, Radiation Protection, Chemistry, Engineering, General Maintenance, Quality Control, and Training departments, as well as contractor personnel.

Completed corrective actions for this event include:

(1) CEDM housings 9 and 13 were removed from the reactor vessel head per Maintenance Work Orders (MWOs) 904996 and 904997 respectively. They were visually inspected and sectioned per MWOs 905030, 905048 and 905069. ABB-Combustion Engineering was contracted to perform detailed destructive and metallurgical examinations of the cracked housings.

Reactor vessel head locations numbers 9 and 13 were capped by CEDM blind flange assemblies under Modification MR-FC-90-74. These assemblies have been analyzed for material compatibility and structural strength in accordance with applicable sections of the ASME Boiler and Pressure Vessel Code. The blind flange assemblies were leak tested during the RCS leak test on January 6, 1991, per Surveillance Test Procedure OP-ST-RC-3007, prior to startup and power operation. No leakage was identified.

(2) Visual inspections of CEDMs 1 and 4 were performed per MWO 905048 to determine if any damage due to steam impingement had occurred. No damage was found. Visual inspections of CEDMs 7, 11, 14, 15, 17, 32, 34, and 38 were performed per MWO 905051 to determine if any cracking was apparent on those housings. No defects were found. A UT examination was then performed on CEDM housings 7 and 11 to detect any cracks that may have been present. No cracking was identified in these housings. An evaluation determined that, due to their self venting feature through mechanical seals, the remaining 37 active CEDM housings are not susceptible to TGSCC.

(3) Operating Instruction Procedure OI-RC-3, "Reactor Coolant System (RCS) Startup" was revised to add a step to vent the HJTC housings after the reactor coolant pumps are started and the reactor coolant pump seals are vented. This will ensure the venting of any air bubbles that may become trapped in the HJTC housings when the reactor coolant pumps are started.

(4) An enhanced Reactor Coolant System leakage action plan has been implemented to provide direction in the event of any future increases in the RCS leakage rate.

The following corrective action will be completed:

An evaluation supporting the unvented blind flange modification for the life of the plant will be completed and provided as backup documentation to the existing analysis which allows use of the unvented blind flange assemblies for the remainder of Cycle 13. The scheduled completion date for this evaluation is June 30, 1991.

There have been no other LERs concerning RCS leakage due to transgranular stress corrosion cracking. LER 84-08 concerned a steam generator tube rupture which was the result of secondary side intergranular stress corrosion cracking, a different mechanism.

OPPD will discuss with Region IV personnel the scope of future inspections deemed necessary. This discussion will occur prior to the 1991 refueling outage.