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

March 6, 1991
LIC-91-0009L

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 91-03 for the Fort Calhoun Station

Please find attached Licensee Event Report 91-031 dated March 6, 1991. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i[4])(B).

If you should have any questions, please contact me.

Sincerely,

J K Gasper

W. G. Gates for
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

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station Unit No. 1	0 5 0 0 0 2 8 5	1 OF 0 7

TITLE (4)

Containment Penetration M-3 Outside Design Basis

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	0 5 0 0 0
0 2	0 4	9 1	9 1	0 0 3	0 0	0 3	0 4	9 1		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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)
D. J. Bannister Shift Technical Advisor	4 0 2 5 3 3 - 6 8 3 1

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
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On February 4, 1991 with the plant operating at 100% power, the Plant Review Committee determined that containment penetration M-3 which is associated with the Chemical Volume Control System (CVCS) was outside the design basis specified in a Safety Evaluation Report (SER) which allowed exclusion from Type C testing required by 10 CFR 50 Appendix J. Past approval of Emergency Operating Procedure requirements and implementation of modifications which changed the post-accident containment pressure profile had invalidated the SER basis.

This condition resulted from failure to incorporate design basis information into the Updated Safety Analysis Report (USAR). Procedural controls have been instituted to assure the design basis for penetration M-3 is met; these controls will be further enhanced in the future. The USAR will be updated to include the design basis for this penetration.

At Fort Calhoun Station Unit No. 1, the Chemical Volume Control System (CVCS) provides a means of maintaining Reactor Coolant System inventory (by charging and letdown), chemistry (by demineralization), purity (also by demineralization), and reactor reactivity control (by boration and dilution) during normal and refueling operations. The CVCS charging line enters containment via penetration M-3. The charging line penetrating containment has no automatic containment isolation valves; however, it does have a single check valve (CH-198) located immediately outside containment which opens in the direction of charging flow. This configuration was approved by the NRC in the original plant Safety Evaluation Report (SER) based on similarity to a previously approved configuration.

During development of the 10 CFR Part 50, Appendix J test program for Fort Calhoun Station, testing requirements for this penetration were extensively reviewed. Omaha Public Power District (OPPD) provided justification to the NRC for not including this penetration in the local leak rate testing program. On January 26, 1983, OPPD formally requested an exemption from Type C testing for isolation valves associated with penetration M-3 using justification as follows. Containment integrity would be maintained due to the high discharge pressures developed by the charging pumps (in excess of 2100 psia) being greater than the maximum post-accident containment pressure (60 psig). During an accident all available charging pumps would be automatically started and their suction realigned to the Concentrated Boric Acid Storage Tanks (CH-11A & CH-11B) upon receipt of a Safety Injection Actuation Signal (SIAS). Thus, the charging pump flow would provide a sealed barrier against any release of radioactivity from containment for approximately 80 minutes. Containment pressure would drop to about 2 psig within 50 minutes of the accident.

On January 10, 1986, the NRC issued an approval document for exemption from certain requirements of Appendix J. The SER included with this document concurred with the OPPD justification for not testing check valve CH-198. It noted that even after the boric acid storage tanks are empty, there would still exist a 14 foot water head on the suction side of the charging pumps providing approximately 6 psig to provide a seal against leakage. Further, the staff found that an exemption from Appendix J was not needed for this check valve since the valve was not included in the valve categories of paragraph II.[illegible] Appendix J, which are required to be Type C tested. Therefore, the valve was to be excluded from the Type C test program.

In January of 1986, Fort Calhoun Station converted from its old event based Emergency Procedures to functional based Emergency Operating Procedures (EOPs). The EOPs implemented required the control room operators at 30 minutes following a LOCA to either swap charging pump suction to the Safety Injection and Refueling Water Storage Tank (SIRWT) (if SIRWT level indication is greater than

72"), or stop the charging pumps (if SIRWT level indication is less than 72"). This would minimize boron precipitation in the reactor core on a cold leg break to support post-LOCA long term core cooling.

Assuming the minimum Technical Specification SIRWT level (186"), it could be possible for the EOP charging pump stop criteria to be satisfied in 30 minutes. Comparing this to the original containment pressure analysis reveals that containment pressure would be 7 psig at 30 minutes, which is higher than the 2 psig specified in the Appendix J SER, and higher than the 6 psig provided by the charging system elevation head.

The first revision to the original containment pressure analysis in Section 14.16 of the Update Safety Analysis Report (USAR) occurred in the July 1989 annual USAR update. The update occurred as a result of analyses performed in 1988 to re-examine the containment pressure response due to Containment Spray (CS) nozzle blockage (as reported in LER 88-08). The revision also reflected implementation of a modification delaying the time when CS pumps are automatically started by the sequencer timers on receipt of a Containment Spray Actuation Signal (CSAS). The latter was due to concerns over a possible load shed of the engineered safeguards electrical buses, when the CS pumps were started concurrent with other Engineered Safety Feature (ESF) equipment (as reported in LERs 88-32 and 88-33). The 1989 USAR analysis shows that post-accident containment pressure would be approximately 20 psig at 30 minutes and approximately 10.5 psig at 50 minutes, both of which are higher than the 2 psig stated in the Appendix J SER.

The most recent reanalysis of post-LOCA containment pressure response was performed in late 1990. The analysis was performed in response to findings related to below design basis containment air cooler performance and single CS pump operation concerns (as reported in LER 90-25). The analysis shows that post-LOCA containment pressure would remain above 6 psig for 24 hours and be in the 20 psig to 40 psig range at 30 to 50 minutes.

The identification of the problem occurred during the compiling of Design Basis Documents, as part of OPPD's Design Basis Reconstitution Program. An open item with containment penetration M-3 had existed since January of 1990. The open item was written to address an apparent discrepancy between the USAR containment isolation design criteria for Reactor Coolant Exposed Systems and the actual configuration of penetration M-3. The item was originally assigned a Category 2 severity level due to missing documentation. In the process of resolving this open item it was discovered that, since the containment post-accident pressure profile had changed, the assumptions in the 1986 Appendix J SER were no longer valid. The open item was immediately changed to Category 1 (potentially reportable).

On February 4, 1991 with the plant operating at 100% power, the Plant Review Committee (PRC) concluded that due to the revisions in the containment post accident pressure analysis, the containment could potentially be pressurized above the remaining head in the system following the securing of the charging pumps. This placed penetration M-3 outside of its design basis as stated in the 1986 Appendix J SER, a reportable condition pursuant to 10 CFR 50.73(a)(2)(ii)(B). A one-hour telephone notification pursuant to 10 CFR 50.72(b)(1) was made on February 4, 1991.

In summary, the original USAR Section 14.16 Containment Pressure Analysis indicated the containment pressure would be reduced to near atmospheric levels (approximately 2 psig) within 50 minutes of the accident, such that containment isolation was assured by the fluid pressure in penetration M-3. The containment pressure analysis and charging pump running time provided the main basis for not subjecting the M-3 penetration to Type C testing, as noted in the SER supporting exception of check valve CH-198 from Appendix J requirements. However, subsequent to the SER issuance in 1986, the EOPs reduced the charging pump run time, and the containment post-accident pressure analysis was revised to reflect changes in the containment cooling system configuration. Combined they resulted in containment potentially being pressurized above the remaining head in the system after charging pumps would be secured. This placed penetration M-3 outside its design basis as stated in the SER.

There are only two credible leakage paths that could occur post-LOCA from the charging system directly to the atmosphere once the charging pumps (CH-1A, 1B, & 1C) are secured. One path would be from the Reactor Coolant System (RCS) to CH-11A & 11B. The effluent would pass through containment penetration M-3 through check valve CH-198, then through one or more charging pump discharge check valves CH-187, CH-188, or CH-189. The effluent would then have to pass through one or more of the idle charging pumps (which are positive displacement pumps). Then the effluent would have to pass through either check valve CH-155 of the gravity feed header or through check valves CH-143 and either CH-129 or CH-130 of the Concentrated Boric Acid Pumps (CH-4A & 4B) on its way to CH-11A and/or CH-11B which are both vented to the Auxiliary Building.

The other path would be from the RCS to the SIRWT. The effluent would again pass through penetration M-3, through CH-198, through CH-187, CH-188, and/or CH-189, then through the idle positive displacement charging pumps. Then the effluent would have to pass through check valve CH-156 and the charging pump suction valve from SIRWT (HCV-218-3) on its way to the SIRWT which is also vented to the Auxiliary Building. Leakage through the Volume Control Tank (CH-14) is not considered credible since its outlet valve (LCV-218-2) is automatically closed upon receipt of a SIAS. Also, CH-14 is not vented to atmosphere during normal operation but instead has a hydrogen overpressure.

Since the Auxiliary Building is maintained at a negative pressure, any effluent from CH-11A, CH-11B, or the SIRWT would be handled by the Auxiliary Building ventilation system. The effluent would be processed by High Efficiency Particulate (HEPA) filters, then monitored for gaseous, particulate, and iodine activity prior to being discharged to atmosphere via the Auxiliary Building stack. Any abnormal increase in stack radiation would be annunciated in the control room, which would require action by the control room operators.

The penetration M-3 was deemed operable based upon the fact that the charging pump discharge check valves (CH-187, 188, and 189) had been rebuilt and verified leak-tight in April of 1990. Furthermore, the check valves are routinely verified to be operable in the reverse direction, because one charging pump is in continuous operation pressurizing the header in excess of 2100 psia and the check valves on the idle pumps prevent reverse flow through them. In addition, the piping upstream of the charging pumps (between each pump's suction isolation valve and its discharge check valve) is depressurized biweekly for the performance of a Preventative Maintenance (PM) procedure to recharge the pump discharge pulsation dampers. The check valves provide leak-tight isolation against the greater than 2100 psia discharge pressure of the running charging pump. Based upon this the three charging pump check valves were considered adequate isolation for penetration M-3. The piping between check valve CH-198 and check valves CH-187, CH-188, and CH-189 was not considered a credible leakage path since it was hydrostatically tested to 3000 psig. The entire system is also monitored for leakage daily by the performance of OP-ST-RC-3001 (RCS Leak Rate Calculation).

Safety Analysis for Operability (SAO) 91-01, Rev. 0 was approved on February 6, 1991 to provide the required justification for continued operation based upon the fact that the operability of penetration M-3 and the integrity of the containment can be assured post-accident on an interim basis through operator actions. The charging pumps' manual discharge isolation valves (CH-190, CH-192 and CH-193) would be closed by local operator actions once charging pump stopping criteria are met. The closing of these valves would be prompted by the same procedural step in the EOPs that requires stopping the charging pumps. These valves are considered leak-tight, since during on-line charging pump maintenance these valves have provided leak free isolation against the pressure of a running charging pump (in excess of 2100 psia). The valves were last used for isolation purposes in November and December 1990. Closure of these three isolation valves would ensure the charging header is isolated and no potential leakage path was available for containment leakage through penetration M-3. In addition, each pump discharge check valve in series would provide redundancy to assure proper post-accident containment isolation.

The provisions of SAO 91-01, Revision 0 were implemented through Operations Memorandum 91-01 as an interim method until the EOPs could be changed. Once the EOPs were revised, Rev. 1 to the SAO was issued to reflect this.

To ensure that no other containment penetrations were affected by the revisions to the containment pressure analysis, a detailed review was conducted of each containment penetration's testing requirements to ensure no problems similar to those encountered with penetration M-3 existed. The review concluded that no other discrepancies existed.

The root cause of this condition was that the design basis for the exclusion of penetration M-3 from Appendix J testing as noted in the SER was not documented in the USAR or other design basis document. This information was not readily available to personnel who formulated the EOP actions, and to those who designed the modifications which changed the containment pressure profile. However, the design basis reconstitution process which revealed this condition was intended in part to assure that appropriate information was incorporated into the design basis documents once the reconstitution process was complete. Thus, the overall cause and corrective actions had been previously identified.

The following corrective actions were completed:

(1) Operations Memorandum 91-01 was issued on February 5, 1991 outlining the concerns with penetration M-3 and requiring closing the charging pump discharge isolation valves (CH-190, CH-192, and CH-193) if charging pumps had to be secured post-LOCA. This memorandum was canceled after implementation of the procedure changes noted in (2) below.

(2) Changes have been made to EOP-03 (LOCA), AOP-22 (Reactor Coolant Leak), and EOP-20 (Functional Recovery) to require closing the charging pump discharge isolation valves (CH-190, CH-192, and CH-193) if the charging pumps must be secured post-LOCA. The basis for the EOP/AOP changes were documented in the Technical Basis Document of the EOPs and AOPs.

(3) SAO 91-01, Rev. 0 and 1 were issued to justify operation based on the provisions for operator actions in Operations Memorandum 91-01 and the subsequent changes to the EOPs and AOPs described in (1) and (2) above.

(4) Other containment penetrations were evaluated which confirmed that the problem with penetration M-3 was an isolated occurrence.

The following additional corrective actions will be taken:

(1) Appropriate EOPs/AOPs will be revised to maintain penetration M-3 at a pressure higher than containment pressure for the duration of a postulated accident. These changes will be made by May 15, 1991.

(2) The USAR will be revised to include the updated containment pressure analysis, and to include the design basis of penetration M-3. This will be included in the 1991 annual update.

As discussed earlier, LERs 88-08, 88-32, and 88-33 reported on conditions that lead to the reanalysis of containment post-accident pressure in 1988. LER 90-25 reported on containment cooling design basis issues that led to the reanalysis of containment post accident pressure in 1990. LERs 87-38 and 88-11 reported on conditions that could have resulted in a potential degradation in containment integrity. LER 88-04 reported on a containment isolation valve that was outside its design basis.