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October 28, 1996

U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318; License No. DPR 69
Licensee Event Report 96-004
Missed Surveillance Due to Less than Adequate Technical Review of Surveillance Test Procedure

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. Should you have questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

Peter Katz

PEK/RCG/dlm

Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
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IE22 1/1

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

APPROVED BY OMB NO. 3150-0104 EXPIRES: 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENISNG PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Calvert Cliffs, Unit 2	05000 318	1 OF 6

TITLE (4)

Missed Surveillance Due to Less than Adequate Technical Review of STP

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
									Calvert Cliffs, U2	05000 318
09	26	96	96	-- 004	-- 00	10	28	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include Area Code)
R. Cary Gradle, Compliance Engineer	410-495-3738

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., approximately15 single-space typewritten lines) (16)

On September 26, 1996, it was identified that the Unit 2 Surveillance Test Procedure (STP) used to verify containment closure in Mode 6 did not meet the requirements of the Technical Specification Surveillance Requirements. During past refueling operations, this STP did not contain a step to verify closure of an auxiliary feedwater drain line manual isolation valve. The missed surveillance was due to less than adequate previous technical reviews of the STP. This event did not result in any significant safety consequences. Corrective action included: (1) procedure revision to include closure verification for the identified valve; (2) historical review of surveillance events to eliminate any programmatic concern; and (3) a complete and thorough technical closure verification of the appropriate Unit 1 and Unit 2 procedures. No other deficiencies were identified.

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Calvert Cliffs, Unit 2	05000 318	96	- 004 -	00	02 OF 06

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

I. DESCRIPTION OF EVENT

On September 26, 1996, a Functional Surveillance Test Coordinator (FSTC) at Calvert Cliffs identified that the Unit 2 Surveillance Test Procedure (STP) used to verify containment closure did not meet the requirements of the Unit 2 Technical Specifications Surveillance Requirements. Specifically, during past refueling operations (MODE 6) when No. 22 Steam Generator was open to the containment atmosphere, STP O-55A-2 (Revision 16), "Containment Closure Verification," (Attachment 2) did not contain a step to verify closure of auxiliary feedwater (AFW) system drain line manual isolation valve (2-AFW-1023). This was contrary to the Technical Specification Surveillance Requirement 4.9.4 which requires, in part, that within 72 hours prior to the start of and at least once per 7 days during core alterations and movement of irradiated fuel in the containment, that each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be verified closed.

A similar closure verification discrepancy for No. 21 Steam Generator did not exist. The corresponding procedure, STP O-55A-1, for Unit 1 was reviewed and no similar deficiencies were identified. At the time of discovery, Unit 2 was at 100 percent power.

Auxiliary feedwater manual isolation drain valve (2-AFW-1023) is piped in series with two other manual isolation valves (2-AFW-1022 and 2-AFW-1021) on the same AFW drain line. All three of these drain valves are normally closed as required by established plant procedure. This drain line taps off the AFW supply header to No. 22 Steam Generator just before the supply header enters the Unit 2 containment. The AFW supply header then passes through a check valve (2-AFW-130) to No. 22 Steam Generator.

Technical Specification Limiting Condition for Operation (LCO) 3.9.4 describes the required status of containment penetrations during refueling operations. Technical Specification 3.9.4 states, in part, that "the containment penetrations shall be in the following status:

b. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Be capable of being closed by an OPERABLE automatic containment purge valve."

Technical Specification 3.9.4 is applicable only during core alterations and movement of irradiated fuel within the containment. The LCO Action Statement specifies that with the requirements of the LCO not satisfied, immediately suspend all operations involving core alterations or movement of irradiated fuel in containment. The bases for Technical Specification 3/4.9.4 states that "during core alterations or movement of irradiated fuel within containment, release of fission product radioactivity to the environment must be minimized."

The primary purpose of STP O-55A-2 is to verify Unit 2 containment closure in MODE 5 (Cold Shutdown) or MODE 6 (Refueling) by ensuring that each containment penetration providing direct access from the containment atmosphere to the outside atmosphere is closed or controlled. Unlike STP O-55-2, "Containment Integrity Verification," [performed monthly when Unit 2 is in MODE 1 (Power Operation) through MODE 4 (Hot Shutdown)], STP O-55A-2 contains additional closure verification requirements for Nos. 21 and 22 Steam Generators. These additional valve closure verifications are conducted by performance of attachments to the procedure. Satisfactory performance of STP O-55A-2 is credited with fulfilling the surveillance requirement of Technical Specification 3.9.4. Surveillance Test Procedure O-55A-2, Section 6.3 (6.4), "21 (22) Steam Generator Checks," verifies the current steam generator status (open or closed to the containment atmosphere) and isolation requirements by performance of the applicable attachment. With No. 22 Steam Generator open to the containment atmosphere, Attachment 2 of STP O-55A-2 (Revision 16) did not contain a step to verify closure of 2-AFW-1023. This verification was required by Technical Specification 3.9.4 to ensure the required isolation configuration for No. 22 Steam Generator.

In 1987, a review of all penetrations that could provide a direct flow path from inside to outside containment was conducted. The intent of this review was to provide increased confidence that a direct path could not be inadvertently created through a penetration that could be overlooked during core alterations. A set of acceptance criteria was established. One of these acceptance criteria was to identify all penetrations that could provide a direct flow path from inside to outside containment and assure that all such penetrations were isolated by at least one valve, blank flange, or other approved barrier.

An investigation of STP O-55A-2 determined that following the 1987 penetration review discussed above, Revision 11 (dated April 10, 1987) of STP O-55A-2 did not require closure verification for 2-AFW-1023 when No. 22 Steam Generator was open to the containment atmosphere during MODE 6. However, Revision 11 did require closure verification for the similar valve (2-AFW-1063) for No. 21 Steam Generator under the same plant conditions. Subsequent revisions to

STP O-55A-2, through Revision 16 (dated May 27, 1992), were reviewed but none required closure verification for 2-AFW-1023. A similar historical review was conducted on STP O-55A-1 (Unit 1) with no similar discrepancy identified.

Immediately upon discovery of the deficient condition of STP O-55A-2, the procedure was recalled to correct the identified discrepancy. Surveillance Test Procedure O-55A-1 was reviewed but no similar error was found.

II. CAUSE OF EVENT

The immediate cause of this event was a procedural deficiency. The procedure, STP O-55A-2 (Attachment 2), did not contain closure verification for 2-AFW-1023, an AFW system drain line manual isolation valve, during refueling operation (MODE 6) when No. 22 Steam Generator was open to the containment atmosphere. The root cause of this event is that previous technical reviews of this infrequently performed procedure attachment were less than adequate to identify and correct the omitted closure verification for 2-AFW-1023. A review of all penetrations that could provide a direct flow path from inside to outside containment during core alterations was conducted in 1987 (discussed above). This review, as well as subsequent technical reviews (e.g., STP biennial reviews, procedure reviews, and revisions), did not identify that Attachment 2 of STP O-55A-2 did not contain closure verification for 2-AFW-1023 as required by Technical Specification Surveillance Requirement 4.9.4.

III. ANALYSIS OF EVENT

The event is considered reportable in accordance with 10 CFR 50.73(a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications." Contrary to the requirements of Unit 2 Technical Specification 3/4.9.4, a containment penetration manual isolation valve was not verified closed. This manual valve (2-AFW-1023) is in series with two other manual valves (2-AFW-1022 and 2-AFW-1021) on the same AFW drain line. All three of these valves are normally shut as required by established plant procedure. In the unlikely event, during refueling operations, that all three of these valves were open in conjunction with No. 22 Steam Generator open to the containment atmosphere (i.e., steam generator manway open) and the AFW header drained upstream of 2-AFW-130 (AFW to No. 22 Steam Generator check valve, located inside containment) then potentially a pathway could have existed from the containment atmosphere to the outside atmosphere.

Technical Specification 3.9.4 requires that the containment penetrations be closed or be capable of being closed by an operable automatic valve, except for the personnel air lock (PAL) doors, during core alterations and movement

of irradiated fuel within the containment. Chapter 14 of the Calvert Cliffs Updated Final Safety Analysis Report evaluates the fuel handling incident and concludes that the 0-2 hour exclusion boundary doses resulting from the fuel handling incident are within the guidelines of 10 CFR Part 100. This Chapter 14 analysis assumes that the entire volume of one containment atmosphere is released to the outside air during this two hour time period. In addition, the analysis assumes that both PAL doors may be open during this time and all activity released from the containment is assumed to be unfiltered. This analysis results in a maximum offsite dose of 14.06 Rem to the thyroid and 0.457 Rem to the whole body. The Standard Review Plan guidelines are 25% of the 10 CFR Part 100 limits, e.g., 75 Rem to the thyroid and 6 Rem to the whole body.

An analysis was performed by our Nuclear Engineering Unit to determine the consequences for both a fuel handling incident and a loss of shutdown cooling if the pathway existed through the AFW drain line discussed previously. The analysis determined that this pathway would contribute less than 0.03 percent to the activity exiting containment from a fuel handling incident and 0.02 percent from a loss of shutdown cooling. In both cases, this analysis concluded that the minute increase associated with this steam generator pathway would not cause the previous very conservative analytical assumptions to be invalidated. Thus, the consequences of any accident previously analyzed did not significantly increase. Based upon this event specific analysis and the fact that this event did not increase the probability of the incidents discussed above, it is concluded that this event did not result in any significant safety consequences.

IV. CORRECTIVE ACTIONS

Immediate

A. STP O-55A-2, "Containment Closure Verification," was recalled. The corresponding Unit 1 procedure, STP O-55A-1, was reviewed but no similar discrepancy was found.

Long-Term

A. To ensure the steam generator isolation requirement verifications are being met by STP O-55A-1 and STP O-55A-2, a senior licensed operator of our Procedure Development and Modification Acceptance Unit conducted a thorough technical verification review of the appropriate sections of these procedures. No other deficiencies were identified.

B. We reviewed the history of surveillance events to identify a pattern. Based upon our review and investigation of this event, we do not consider the omitted closure verification for valve 2-AFW-1023 in STP O-55A-2 to be indicative of a programmatic concern.

C. STP O-55A-2 has been revised, effective date of October 7, 1996, to include the appropriate closure verification for 2-AFW-1023.

V. ADDITIONAL INFORMATION

A. Affected Component Identification:

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Containment	N/A	N/A
Steam Generator	SG	N/A
Auxiliary Feedwater	N/A	BA
Isolation Valve	ISV	BA

B. Previous Similar Events:

Since the completion of our STP program upgrade and technical adequacy review project in December 1992, we have reported three missed surveillance requirements in LERs (317/93-002, 317/93-006 and 317/94-002), none of which related to less than adequate technical review of the STP. The STP program technical adequacy review, mentioned above, was discussed as corrective action in LERs (317/89-013, 317/89-017, 317/89-024, 317/90-001, 317/90-07, 317/90-08, 317/90-10, 317-90-15, 317/91-02 and 317/91-05).