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Constellation Nuclear

Calvert Cliffs
Nuclear Power Plant

A Member of the Constellation Energy Group

August 30, 2001

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 2001-002
Misunderstood Technical Specifications Causes Containment Closure Deviation

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 E. Katz

PEK/TWG/bjd

Attachment

cc: R. S. Fleishman, Esquire
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NRC FORM 366 (6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

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	050000 318	1 OF 009

TITLE (4)
Misunderstood Technical Specifications Causes Containment Closure Deviation

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
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OPERATING MODE (9)	6	THIS REPORT IS SUBMITTED PURSUANT O 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)
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			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)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T. W. Grover	410-495-2064

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (16)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On July 2, 2001, it was discovered that the shutdown cooling pumps were removed from operation for local leak rate testing of containment penetration 41 with the normal containment sump valves open during the 2001 Calvert Cliffs Unit 2 refueling outage. This condition is not allowed by Technical Specification (TS) 3.9.4. The plant was in Mode 6 with the Reactor Coolant System (RCS) at 105 degrees Fahrenheit and atmospheric pressure, refueling pool water level was greater than 23 feet above the fuel in the reactor vessel, core alterations had been suspended, no operations were permitted that would cause a reduction to RCS boron concentration, and a spent fuel pool cooling loop was providing RCS decay heat removal. The root cause of this event was that TS containment penetration closure requirements were not accurately understood and implemented. These requirements have now been properly implemented in procedures. The inaccurate TS implementation was limited to the normal containment sump valves. A review of all deleted TS interpretations is underway to ensure that TS requirements are being accurately implemented. The safety significance of this event is very low.

I. DESCRIPTION OF EVENT

On July 2, 2001, it was discovered that the shutdown cooling (SDC) pumps were removed from operation for local leak rate testing (LLRT) of SDC return isolation motor-operated valves (MOVs), 2-SI-651 and 2-SI-652 (Containment Penetration 41), with the normal containment sump outlet valves, 2-MOV-5462 and 2-MOV-5463, open during the 2001 Calvert Cliffs Unit 2 refueling outage. This condition is not allowed by the Limiting Condition for Operation (LCO) in Technical Specification (TS) 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level." At the time of the condition, the plant was in Mode 6 with the Reactor Coolant System (RCS) at 105 degrees Fahrenheit and atmospheric pressure, refueling pool water level was greater than 23 feet above the fuel in the reactor vessel, core alterations had been suspended, no operations were being permitted that would cause a reduction to RCS boron concentration, and a spent fuel pool cooling loop was providing RCS decay heat removal.

Note 2 in TS LCO 3.9.4 allows the SDC loops not to be in operation during the time required for LLRT of containment penetration 41 pursuant to the requirements of Surveillance Requirement 3.6.1.1 or to permit maintenance on valves located in the common SDC suction line, provided:

a. no operations are permitted that would cause a reduction to RCS boron concentration,

b. CORE ALTERATIONS are suspended, and

c. all containment penetrations are in the status described in LCO 3.9.3.

This note is applicable during Mode 6 with the water level greater than or equal to 23 feet above the top of the irradiated fuel assemblies seated in the reactor vessel.

Technical Specification LCO 3.9.3 requires, in part, each containment penetration providing direct access from the containment atmosphere to the outside atmosphere be closed by a manual or automatic isolation valve, blind flange, or equivalent. The TS 3.9.4 Action Statement requires, with one required SDC loop inoperable or not in operation, action to be initiated immediately to restore the SDC loop to operable status and operation.

On April 4, 2001, at 1250, TS LCO 3.9.4 Note 2 was invoked prior to securing SDC flow for LLRT and maintenance on the common SDC suction line. Operations personnel verified no core alterations were in progress, no containment closure deviations existed, no operations that could cause a boron reduction of the RCS were in progress, and the refueling pool water level was greater than 23 feet above the top of the fuel in the reactor vessel. At 1255, the Control Room Operator secured SDC and verified Spent Fuel Pool Cooling Train No. 12 was providing cooling to the refueling pool and the reactor core.

On April 4, 2001, between 1250 and 1800, Operations personnel began draining and pressurizing the SDC header to support LLRT on Containment Penetration 41. While draining the SDC header, the normal containment sump became full and a normal containment sump high-level alarm was received in the Control Room. Draining the SDC header could not continue until the water in the normal containment sump was removed. Operations personnel reviewed the options available to remove the water from the normal containment sump. One option was to pump the water from the normal containment sump to a temporary storage container set up inside Containment. The other option was to continue to drain the SDC header to the normal containment sump and open the normal containment sump outlet valves upon receipt of a high level alarm. Operations personnel reviewed Nuclear Operations (NO)-1-114, "Containment Closure" and Surveillance Test Procedure (STP) O-55A-2, "Containment Closure Verification" to determine if the containment normal containment sump outlet valves could be opened, when containment closure conditions were in effect. Operations personnel concluded that the procedures allowed the normal containment sump outlet valves to be open provided the spring return handswitch on 2-MOV-5463 was not key-overridden open. For the next 4 to 5 hours, upon a high level alarm, Operations personnel verified the spring return handswitch on 2-MOV-5463 was not key-overridden open and opened the normal containment sump outlet valves (2-MOV-5462 and 2-MOV-5463) for approximately 15 seconds to drain the normal containment sump.

On April 4, 2001, at 1800, the day shift Operations crew turned over to the night shift Operations crew. The night shift Operations crew continued to open the normal containment sump outlet valves (2-MOV-5462 and 2-MOV-5463) to drain the normal containment sump upon a high level alarm in accordance with plant procedures in support of draining the SDC suction line for LLRT on Containment Penetration 41. Between 1800 and 2245 hours, the Control Room Supervisor questioned whether opening the normal containment sump valves violated containment closure restrictions. The Shift Manager and Control Room Supervisor reviewed NO-1-114 and STP O-55A-2. They concluded that plant procedures allowed the normal containment sump outlet valves to be open in all containment closure conditions, except during core alterations or movement of irradiated fuel, provided the spring return handswitch on 2-MOV-5463 was not key-overridden open. Plant operations personnel also considered the normal containment sump outlet valves would be opened for 15 seconds compared with a time to boil of 10 hours. Draining the normal containment sump via the outlet MOVs continued until the SDC suction line was drained at approximately 2245.

It has been conservatively estimated that outlet MOVs 5462 and 5463 were open approximately 24 times for a total of approximately 6 minutes during a period when TSs 3.9.3 and 3.9.4 required the normal containment sump penetration to be closed. This allowed a path from the containment atmosphere to the outside atmosphere when both SDC loops were inoperable. With LCO 3.9.4 not met, the required action in TS 3.9.4 Action a, for one required SDC loop inoperable or not in operation, to immediately initiate action to restore the SDC loop to operable status and operation was not met.

II. CAUSE OF EVENT

Plant personnel used guidance in STP O-55A and NO-1-114 that conflicted with TS requirements to open the MOVs.

The procedure deficiencies were caused by a misunderstanding of containment closure requirements in the old Calvert Cliffs Standard TS 3.9.4 and 3.9.8.1.a (currently improved TS 3.9.3 and 3.9.4, respectively). Plant personnel thought the statement "all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are maintained closed" meant that all containment penetrations allowing an unfiltered release pathway from the containment atmosphere to the outside environment must be maintained closed. Since the flowpath from the normal containment sump is to the Emergency Core Cooling System pump room, which has a filtered ventilation system, plant personnel thought the normal containment sump outlet valves (2-MOV-5462 and 2-MOV-5463) could be opened in all containment closure conditions. This was incorporated into NO-1-114, Revision 0 on March 13, 1996, and STP O-55A, Revision 16, Change 23 on March 14, 1996. Therefore, the root cause of this event is old TS requirements were not accurately understood and implemented.

A TS interpretation (Serial Number 96-001) was approved on April 25, 1996 as an aid to plant staff to more fully understand old TS 3.9.8.1.a. The TS interpretation reinforced the misunderstood meaning of containment closure requirements in old TS 3.9.8.1.a by interpreting "outside atmosphere" to mean "an unfiltered release to the outside environment." The TS interpretation was not accurate. Therefore, a contributing cause of this event is Licensing staff did not accurately determine the intent of the TS requirement.

Technical Specification interpretation 96-001 was deleted on December 13, 1996 during an effort to eliminate all TS interpretations. Operations staff deleted the TS interpretation, but did not change the affected procedures because the deletion of the TS interpretation did not change the understood meaning of the TS requirements. The mindset at the time was that TS interpretation 96-001 was no longer needed because this information was adequately captured in NO-1-114 and STP O-55A. It appears that this mindset was established based on the following:

A. Plant personnel thought they understood the TS requirement;

B. The understood meaning was already being used in procedures;

C. Plant personnel thought that the TS Bases made it clear that "outside atmosphere" meant an unfiltered release to the outside environment;

D. CEOG-115 (TSTF-197) proposed this same clarification; and

E. A TS change to incorporate CEOG-115 (TSTF-197) was submitted to the Nuclear Regulatory Commission (NRC) earlier that month on December 4, 1996.

Calvert Cliffs requested a change to the TSs to incorporate CEOG-115 (TSTF-197) in the License Amendment Request to convert to the improved TS. In a letter from the NRC to Baltimore Gas and Electric Company, dated May 16, 1997, the NRC withheld accepting this change pending NRC staff approval of CEOG-115 (TSTF-197). Based on comments received from the NRC during review of CEOG-115 (TSTF-197), the change from "outside atmosphere" to environment was eliminated from the traveler. On August 19, 1997, Revision 4 of the License Amendment Request to convert to the improved TS withdrew our changes associated with TSTF-197. It is likely that some Operations personnel were aware of the withdrawn TS change associated with TSTF-197. However, the impact on plant procedures does not appear to have been considered. Therefore, the second contributing cause of this event is existing conditions adverse to quality were not recognized and communicated such that they could be resolved under the corrective action program. Resolution would have likely involved a TS Bases change that would have been implemented in affected procedures.

Several procedure revisions to STP O-55A and NO-1-114 were reviewed and approved subsequent to withdrawing the associated TS changes on August 19, 1997. The technical reviews of these procedure revisions were missed opportunities to question and obtain an accurate understanding of the TS requirement. This also was a contributing cause.

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 TS LCO 3.9.4 Note 2, the normal containment sump penetration (Penetration 8) was not in the status described in LCO 3.9.3, "Containment Penetrations" during a period when both SDC loops were inoperable in Mode 6 with the water level greater than or equal to 23 feet above the top of the irradiated fuel assemblies seated in reactor vessel. The normal containment sump penetration provides direct access from the containment atmosphere to the outside atmosphere and was not closed by a manual or automatic isolation valve, blind flange, or equivalent. With the conditions of the LCO not met, the required action in TS 3.9.4 Action a, for "One required SDC loop inoperable or not in operation" to "initiate action to restore SDC loop to operable status and operation immediately" was not performed.

Technical Specification LCO 3.9.3 is applicable during core alterations and movement of irradiated fuel within Containment. In addition, all containment penetrations must be in the status described in LCO 3.9.3 when the SDC pumps are removed from operation per LCO 3.9.4 Note 2. Technical Specification 3.9.3 requires all containment penetrations be closed with the following exceptions. The containment outage door and the personnel air lock can be open but must be capable of being closed under administrative control. Penetrations providing direct access from the containment atmosphere to the outside atmosphere can be open but must be capable of being closed by an

operable containment purge isolation system. When LCO 3.9.3 requirements are met, a release of fission product radioactivity within Containment will be restricted from escaping to the environment.

Calvert Cliffs philosophy regarding core alterations would not allow handling of irradiated fuel within Containment when normal SDC is inoperable. However, in the very unlikely event that irradiated fuel could have been moved within Containment when the normal containment sump outlet valves were open, the consequences of a fuel handling incident are considered. 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. The fuel handling incident analysis assumes both personnel air lock doors and the containment outage door are open and the unfiltered release of the entire volume of one containment atmosphere to the outside air during this 2 hour time period. 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 percent of the 10 CFR Part 100 limits, i.e., 75 Rem to the thyroid and 6 Rem to the whole body.

An analysis was performed by our Nuclear Engineering Unit to determine the site boundary dose for a loss of SDC event 5 days after shutdown resulting in a 2-hour release of fission product activity within Containment. The analysis assumes the containment leak rate is equal to the boil-off rate of the RCS. In addition, the analysis assumes that all activity released from the Containment is unfiltered. This analysis results in a maximum offsite dose of 0.966 Rem to the thyroid and 6.9E-4 Rem to the whole body.

Our Nuclear Engineering Unit also determined the consequences for both a fuel handling incident and a loss of SDC if the pathway existed through the normal containment sump drain line. The analysis determined that this pathway would contribute less than 0.03 percent to the activity exiting Containment from a fuel handling incident or a loss of decay heat removal event. In both cases, this analysis concluded that the minute increase associated with this normal containment sump 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.

Risk Evaluation:

The probabilistic safety significance of this event was evaluated given the following:

A. No core alterations or fuel movements were in progress.

B. 2-MOV-5462 and 5463 were open for a total of 6 minutes - (24 times for 15 seconds each time)

C. The normal SDC loops were out-of-service.

D. 2-MOV-5463's handswitch spring returns to closed when released.

The core damage frequency increase due to this event is essentially zero; the normal containment sump outlet valves can only impact a mitigating system by draining the RCS inventory into the Auxiliary Building. This can only occur if the RCS inventory were inadvertently drained down to the containment floor. The probability of this draindown, due to human error or a pipe break, is extremely small for a 6-minute period. This very small probability times the probability that the MOVs fail to close is considered to be negligible and essentially zero.

Using very conservative assumptions, the Large Early Release Frequency increased by 3.5E-08 due to this event. The probabilistic risk assessment analysis assumes that fuel damage occurs if the spent fuel pool cooling system fails. This is extremely conservative because the time to boil was 10 hours, which provides many hours to restore decay heat removal. A very conservative 5 percent failure probability of the spent fuel pool cooling system over a 24-hour period was assumed. In addition, the analysis assumes the operators do not attempt to close the MOV that does not spring return to closed upon release of the handswitch (2-MOV-5462). Using these risk insights in conjunction with the NRC's Significance Determination Process, this event was concluded to have had very low safety significance.

Based upon this event specific analysis and the fact that this event did not increase the probability of a fuel handling incident or a loss of SDC event, it is concluded that this event did not result in any significant safety consequences.

IV. CORRECTIVE ACTIONS

A. STP O-55A-1, STP O-55A-2, and NO-1-114 have been revised to accurately implement the containment closure requirements in TS 3.9.4.

B. Operating Instructions (OI)-17D, "Miscellaneous Waste Processing System" has been revised to accurately implement the containment closure requirements in TS 3.9.4.

C. In April of 1997, additional guidance , based on NRC Inspection Manual Part 9900, "Licensee Technical Specification Interpretations," was developed to ensure implementation of the TS requirements does not change the intent or contradict any TS. This guidance has been incorporated into RM-1-104, "Updating the Safety Analysis Report (UFSAR, USAR) and the Technical Specification Bases (TSB)."

D. We are currently reviewing all deleted TS interpretations to ensure that TS requirements are being accurately implemented.

E. The TS Bases will be revised to clarify the meaning of "outside atmosphere."

F. We will conduct training to ensure Operations personnel understand the containment closure requirements in TS 3.9.3 and TS 3.9.4.

V. ADDITIONAL INFORMATION

A. Component Identification

Component	IEEE 803 EIIS Function	IEEE 805 System ID
High Level Alarm	LA	WH
Containment Building	RB	NH
Handswitch	HS	WH
MOV	ISV	WH
Spent Fuel Pool Cooling System	TM	DA
Shutdown Cooling System	TM	BP
Normal Containment Sump	P	WH
Containment Penetration	PEN	NH

B. Previous Similar Events

There have been two similar events due to a failure to accurately implement a plant TS in the past 5 years. These events are described in Licensee Event Report (LER) 318/96004 and LER 318/97002.

The cause of the event described in LER 318/96004 was inadequate review to identify all containment penetrations that provide a direct flow path from containment atmosphere to outside atmosphere. Although this was a similar event the corrective actions could not have prevented the event described in this LER. The inadequate review did not involve an interpretation or misunderstanding of TS requirements.

The cause of the event described in LER 317/97002 was that plant personnel thought that the statement, "close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere," meant, "close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere." The corrective action to conduct a review to identify and correct any similar instances where we may not meet the literal requirements of the plant's TSs resulted in a number of LERs being submitted, but did not identify the misunderstanding of the TS requirements which led to the event described in this LER. This review most likely relied on the misunderstanding of the TS requirements which led to this event. The misunderstanding was not resolved because information from the NRC was not captured in our corrective action program. The corrective action of converting our standard TS to the new improved TS has clarified many of the requirements concerning the required status of containment penetrations to help assure future compliance with TS requirements, but did not resolve the misunderstanding of the TS requirements which led to the event described in this LER. The improved TS Bases were not revised to clarify the meaning of "direct access from the containment atmosphere to the outside atmosphere"

because a condition adverse to quality was not recognized and communicated such that it could be resolved under the corrective action program.

Resolution would have likely involved a TS Bases change that would have been implemented in affected procedures. The corrective action in this LER to revise the applicable TS Bases will ensure the intent of the previous corrective action is met.