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June 13, 2002

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12
LICENSEE EVENT REPORT (LER 2002-002-00)
INCORRECT VALUE IN FUEL HANDLING ACCIDENT ANALYSIS

Attached is Licensee Event Report (LER) No. 2002-001-00, for the Virgil C. Summer Nuclear Station (VCSNS). The report describes the discovery that an incorrect value was used to quantify the environmental release following a postulated Fuel Handling Accident (FHA) Inside Containment. This report is being submitted in accordance with the requirements of 10 CFR 50.73(A)(2)(v).

Should you have any questions, please call Mr. Mel Browne at (803) 345-4141.

Very truly yours,

Stephen A. Byrne

PAR/SAB
Attachment

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File (818.07)
DMS (RC-02-0107)

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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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Virgil C. Summer Nuclear Station	05000395	1 OF 5

4. TITLE
Incorrect Value in Fuel Handling Accident Analysis

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	DOCKET NUMBER
04	18	02	2002	- 002	- 00	06	13	02	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

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)		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)	X	50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)	X	50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		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)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
M. N. Browne, Mgr., Nuclear Licensing & Operating Experience	(803) 345-4141

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
	VI	NA	NA	No					

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 April 18, 2002, while performing an evaluation on control room habitability after a Fuel Handling Accident (FHA) for a NRC request for information related to the Spent Fuel Pool Expansion Project, it was discovered that a prior evaluation used a non-conservative input in the calculation.

The prior evaluation concluded the resulting activity release from a postulated FHA could be rapidly detected and contained within the reactor containment via rapid closure of the purge supply and exhaust isolation valves. This conclusion was based, in part, on the minimum transit time within the exhaust ventilation ducting from the reactor cavity to the purge isolation valves. The transit time has now been determined to be less than the credited time to close the isolation valves, thus creating the potential for an environmental release.

The cause was determined to be an assumption that the FHA occurred in the vicinity of the reactor cavity whereas the new calculation considered more limiting locations within the refueling canal.

Corrective actions were taken to establish administrative controls to restrict operation of the reactor cavity and refueling canal surface ventilation system during periods of core alterations and fuel movement inside containment. This action is expected to have little or no impact on ALARA since plant operating experience has indicated that airborne contamination being emitted from the cavity/canal water surface is not significant.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2001	-- 002 --	00	2 OF 5

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

PLANT IDENTIFICATION

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION

VL Reactor Building Purge Supply and Exhaust System XVB 0001A/B, XVB 0002A/B

IDENTIFICATION OF EVENT

Prior evaluations of a Fuel Handling Accident (FHA) Inside Containment concluded that the resulting activity release can be rapidly detected and contained within the reactor building via rapid closure of the purge supply and exhaust isolation valves. This conclusion was based, in part, on the minimum transit time within the exhaust ventilation ducting from the reactor cavity and refueling canal to the purge isolation valves. The transit time has now been determined to be less than the time credited to close the isolation valves, thus creating the potential for an environmental release. This was identified in Condition Event Report CER-02-1005.

EVENT DATE

April 18, 2002

REPORT DATE

June 13, 2002

CONDITIONS PRIOR TO EVENT

Mode 1, 100% power

DESCRIPTION OF EVENT

Prior evaluations of a Fuel Handling Accident (FHA) Inside Containment concluded that the activity release could be rapidly detected and contained within the reactor building via rapid closure of the purge supply and exhaust isolation valves. This conclusion was based, in part, on the minimum transit time within the exhaust ventilation ducting from

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2002	-- 002	00	3 OF 5

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

the reactor cavity and refueling canal to the purge isolation valves. The minimum transit time from the point at which the activity enters the exhaust duct until it passes the innermost isolation valve has now been determined to be less than the time credited to close the isolation valves, thus creating the potential for an environmental release. As a result, this condition could have prevented fulfillment of a safety function to control the release of radioactive material and thus challenged the plants ability to limit control room dose to within GDC-19 limits and offsite dose to within the acceptance criteria of Standard Review Plan 15.7.4 had a FHA Inside Containment occurred under standard safety analysis assumptions. The offsite dose criteria of 10 CFR 100 was not of issue since existing analyses show that these limits are met without credit for the containment of any failed fuel activity within the Reactor Building.

CAUSE OF EVENT

The initial plant licensing assessment of the transit time from the inlet register of the reactor cavity and refueling canal ventilation system to the isolation valves on the reactor building purge/exhaust ventilation system was based on information presented in response to NRC Question 311.9 and 311.31. Calculations now indicate that the minimum transit time, under worst-case conditions, can be shorter than the value previously evaluated.

The initial assessment of the transit time assumed the FHA occurred in the vicinity of the reactor vessel where a FHA would result in the maximum number of failed fuel rods. The new calculation considered more limiting locations within the refueling canal with respect to transit time, but with less potential fuel rod failures. The cause of this condition is thus attributed to failure to consider the most limiting location (i.e., minimum transit time) for the postulated FHA.

ANALYSIS OF EVENT

As discussed in Section 9.4.8 of the FSAR, a surface ventilation system for the refueling canal and refueling cavity is normally operated during refueling to reduce potential operator exposure. When in operation, air from the refueling canal/cavity enters one of the 17 registers adjacent to the east side of the refueling canal/cavity and is directed to the Reactor Building Purge Exhaust. As discussed in SCE&G's response to NRC Question 311.31, the transit time for the flow of air from the exhaust inlet to the first isolation valve was initially evaluated to demonstrate that activity released from a postulated FHA Inside Containment would be confined to the Reactor Building via rapid detection using radiation monitors RM-G17A/B and rapid closure of the purge isolation valves. The conclusion of "no release" was based on an instrument channel response time of 100 milliseconds, a transit time for the air from the refueling canal exhaust inlet to the first RB purge isolation valve of 3.84 seconds, and a purge isolation valve closure time of 3 seconds (i.e., 2 seconds to close plus one second to fully seat). The transit time was derived from layout information presented in response to NRC question 311.9. More recent calculations show that the air transit time can be as low as 2.38 seconds; thus creating the potential for a rapid release when transport delays from the water surface to the exhaust inlet are ignored and standard safety analysis assumptions are employed. This indicates that the prior assessment did not address worst-case conditions.

As outlined in NRC Question 311.19 and recent NRC guidance (draft RG-1113), the release of a small fraction of the source term from a FHA is permissible so long as the resulting control room and offsite doses are acceptable. Consequently, to preclude the potential of a rapid activity release, operation of the surface ventilation system for the refueling canal/cavity is now restricted during periods of core alterations and fuel movement inside containment. Given that the activity release from a FHA must now mix with the RB atmosphere and that means to rapidly detect and isolate the purge isolation valves is maintained, the release will be sufficiently restricted to ensure that the resulting doses do not represent more than a minimal increase in the consequences of FHA previously evaluated. This administrative control is expected to have little or no impact on ALARA since plant-operating experience over the past 12 refueling outages has indicted that airborne contamination being emitted from the surface of the canal/cavity is not significant.

IMPACT ON PRIOR PLANT OPERATION

An inspection of prior test results for the purge supply/exhaust isolation valves shows that the valves have always met the previously assumed isolation time of 3.0 seconds. Measured valve performance also indicates that the containment isolation function (i.e., closure of one of the two in-series isolation valves) would actually have been achieved in $\leq$2.5 seconds upon receipt of a valve closure signal. With an instrument response time of 100 milliseconds (i.e., for detection of high radiation by RMA-17A/B) and the worst-case transit time of 2.38 seconds within the exhaust ducting, the actual time available for activity to escape from the Reactor Building has been on the order of 0.22 seconds or less.

Several factors have been neglected in defining the available release period that, if considered, would either offset the 0.22-second release period or minimize the release. For example, all of the activity from the failed fuel will not reach the pool surface at the same instant in time. Therefore, only a fraction of available activity could be released from the Reactor Building. In addition, a high radiation signal would be expected prior to the activity breaking the pool surface, and time is required for the activity to exit the pool and enter the exhaust ducting. Therefore, with these additional time delays, the purge supply/ exhaust isolation valves would likely close prior to any release of activity from the Reactor Building. Any activity that should be released would also pass through the, non-safety, purge exhaust charcoal filters, which would significantly reduce the iodine activity release to the environment with a corresponding reduction in the limiting thyroid doses (DF = 20) following the postulated FHA Inside Containment.

Given the above, there is reasonable assurance that little or no activity would have been actually released to the environment had a FHA Inside Containment occurred and that the resulting doses, both offsite and within the control room, would have been minimal.

INTERIM CORRECTIVE ACTIONS

None

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2002	-- 002	00	5 OF 5

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

OTHER CORRECTIVE ACTIONS

Upon completion of the evaluation of this issue, administrative controls were implemented to restrict operation of the reactor cavity and refueling canal surface ventilation system during periods of core alteration and fuel movement inside containment. This was captured in Surveillance Test Procedure STP 110.001, Pre-Core Alteration Verifications

The surveillance test procedure (STP 0130.05B) that governs the performance of the stroke time test for the Reactor Building purge supply and exhaust isolation valves will be revised to assure that the analysis assumption for closing time is maintained.

PRIOR OCCURRANCES

None