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Gary J. Taylor
Vice President
Nuclear Operations

June 13, 1996
RC-96-0143

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 960004)

Attached is Licensee Event Report No. 96-004 for the Virgil C. Summer Nuclear Station. This report is submitted pursuant to the requirements of 10CFR50.73(a)(2)(ii).

Should you have questions, please call Mr. Jim Turkett at (803) 345-4047.

Very truly yours,

Gary J. Taylor

JWT/GJT/nkk
Attachment

c: J. L. Skolds
W. F. Conway
R. R. Mahan (w/o attachment)
R. J. White
S. D. Ebneter
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S. R. Hunt
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STATE OF SOUTH CAROLINA :

: TO WIT :

COUNTY OF FAIRFIELD :

I hereby certify that on the 13th day of June 1996, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Gary J. Taylor, being duly sworn, and states that he is Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Jeffrey W. Pease
Notary Public

My Commission Expires

July 26, 2005
Date

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

LICENSEE EVENT REPORT (LER)

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TITLE (4)

Unanalyzed Condition Regarding Reactor Building Line Break Analyses

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
										05000
0 5	13	96	9 6 --	04 --	0 0	06	13	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	0	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	X 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 or in NRC Form 366A
		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)
George A. Lippard, III Manager, Nuclear Licensing & Operating Experience	(803) 345-4810

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	BA			N					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE	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)

During Integrated Safeguards Testing, Steam Generator (SG) water levels were observed increasing with the motor-driven Emergency Feedwater (EF) pumps in operation, even though the EF flow control valves (FCVs) were closed. On May 13, 1996, it was determined that this condition was not covered by the Emergency Operating Procedures (EOPs) and may not be addressed in the safety analysis report.

The EF FCV leakage could impact the safety analysis in three areas: 1) the reactor building (RB) flood level for a steamline break (SLB) inside containment, 2) long term RB pressure response for a postulated SLB inside containment and 3) EF flow diversion from the intact SGs during a postulated feedwater line break.

Any potential adverse effect to the plant would have been restricted to an event requiring EF isolation. The cause of this condition is attributed to normal degradation of the valve seats. Engineering analysis has determined that, with operator action by 30 minutes into the event, manual isolation of the EF FCV leakage will assure the established safety criteria are met. Applicable EOPs have been revised to require operator action to close manual valves located upstream and downstream of the FCVs to assure EF flow to a faulted SG is terminated. The leaking valves will be reworked by the end of the next refueling currently planned for the Fall of 1997.

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

PLANT IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION:

Motor-Driven Emergency Feedwater Pump Flow Control Valves IFV03531-EF, IFV03541-EF, IFV03551-EF
Turbine-Driven Emergency Feedwater Pump Flow Control Valves IFV03536-EF, IFV03546-EF, IFV03556-EF
EIIS System Code - BA

IDENTIFICATION OF EVENT:

During Integrated Safeguards Testing, steam generator levels were observed increasing as a result of leakage past the Motor-Driven Emergency Feedwater Pump Flow Control Valves. Upon further investigation on May 13, 1996, it was determined that flow past these valves may result in an unanalyzed condition associated with the reactor building line break analyses. This concern was expanded to include the Turbine-Driven Emergency Feedwater Pump Flow Control Valves.

EVENT DATE:

May 13, 1996

REPORT DATE:

June 13, 1996

This report was generated by Off-Normal Occurrence Report 960243.

CONDITIONS PRIOR TO THE EVENT

MODE 5 0% Reactor Power

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DESCRIPTION OF EVENT

On May 13, 1996, plant personnel determined that leakage past the flow control valves of the Emergency Feedwater (EF) Pumps has the potential for placing the plant in an unanalyzed condition in regards to the Reactor Building line break analyses. The plant was in a refueling outage (Mode 5) conducting Integrated Safeguards Testing when the leakage was detected through observation of increasing steam generator water levels.

CAUSE OF EVENT:

The cause of the event is attributed to normal degradation of the valve seats. The valves are used to adjust feedwater flow to the steam generators during start-up and shutdown. These valves have been in service since the system was constructed and have not required any work to be performed on their seats prior to the event.

Additionally, valve operator air pressure was determined to have an influence on the amount of leakage by not maximizing seating thrust during valve closure. Air operator pressure for the valves was adjusted, as necessary, to provide additional seat loading. Additional testing of the motor-driven EF pump lines was performed to further quantify leakage through the flow control valves when closed and to determine if flow leakage could be isolated by closing the upstream or downstream manual isolation valves associated with the flow control valves. This additional testing confirmed the flow control valves were still leaking in the closed position, but that flow was terminated when either manual isolation valve was closed.

ANALYSIS OF EVENT:

The EF flow control system is designed to isolate flow to a faulted steam generator to limit the mass input to containment, and to terminate the diversion of flow from the intact steam generators. Under certain specific single failure scenarios, the analysis did not address the potential for leakage past a closed EF flow control valve.

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ANALYSIS OF EVENT: (Cont'd)

A leaking flow control valve could impact the plant safety analyses in three areas: 1) the reactor building flood level for a steam line break inside containment, 2) long term reactor building pressure response for a postulated steam line break inside containment and 3) emergency feedwater flow diversion from the intact steam generators during a postulated line break. Supplemental analyses of these three areas had to address the leakage from initiation of the event until the operator manually isolates emergency feedwater flow to the faulted steam generator within 30 minutes. To assure that these additional analyses encompassed any additional leakage, a conservative value of 50 GPM was established as the maximum leakrate through any flow control valve.

The conclusions of these analyses are that an additional 50 GPM emergency feedwater flow control valve leakage is acceptable to assure that the plant is maintained within its design basis provided that flow control valve leakage is terminated within 30 minutes from the start of the event.

IMMEDIATE CORRECTIVE ACTIONS:

Following identification of the concern on May 13, 1996, additional testing was performed to further quantify the worst case leakage and to establish if closure of the upstream and/or downstream manual isolation valves would stop the leakage. The leakage was terminated successfully by manual isolation, so the applicable Emergency Operating Procedures (EOPs) were revised to require operator action to manually close the emergency feedwater isolation valves to a faulted steam generator. Validation of the EOPs was performed on the Control Room Simulator and through in-plant walkdown. Engineering reviews were performed on the leakage rate data to confirm that the leakage was much less than 50 GPM and would meet analysis assumptions provided appropriate operator action was taken.

Based on the conclusions of the engineering analyses and the revisions to the applicable emergency operating procedures, continued operation does not present an adverse impact on safety.

ADDITIONAL CORRECTIVE ACTIONS

The following additional action is planned:

- Rework leaking EF flow control valves prior to restart from the next refueling (RF10) currently scheduled for Fall 1997.

PRIOR OCCURRENCES:

None