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SOUTHERN COMPANY
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April 21, 1997

Docket No. 50-321

HL-5373

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Rod Block Monitor Operability Requirements
Do Not Match Rod Withdrawal Error Analysis Assumptions

Gentlemen:

In accordance with the requirements of 50.73(a)(2)(v), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning rod block monitor Technical Specifications operability requirements that did not match rod withdrawal error analysis assumptions.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

OCV/eb

Enclosure: LER 50-321/1997-003

IE22 1/1

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
NORMS

290040

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

NRC FORM 366 (5-92) | U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB7714), 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)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Edwin I. Hatch Nuclear Plant - Unit 1 | 05000321 | 1 OF 5 |

TITLE (4)
RBM Operability Requirements Do Not Match Rod Withdrawal Error Analysis Assumptions

| 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(S) |
| 03 | 25 | 97 | 97 | 003 | 00 | 04 | 21 | 97 | Plant Hatch - Unit 2 | 05000366 |
| | | | | | | | | | | 05000 |

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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) | X | 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) | | 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 (include area code) |
|---|---|
| Steven B. Tipps, Nuclear Safety & Compliance Manager, Hatch | AREA CODE 912 367-7851 |

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 15 single-space typewritten lines) (16)

On 3/25/97 at 1307 EST, Unit 1 was in the Run mode at a power level of 2558 CMWT (100 percent rated thermal power) and Unit 2 was in the Refuel mode with the cavity flooded. At that time, plant personnel determined an event reported to Southern Nuclear Operating Company personnel by General Electric involving the Rod Block Monitor (RBM) met the reporting requirements of 10 CFR 50.72 (b)(2)(iii) and 50.73 (a)(2)(v). General Electric reported that their core design procedures contained a step which allowed them to assume the RBM is operable and available to mitigate the consequences of a Rod Withdrawal Error (RWE) irrespective of the individual plant's Technical Specifications requirements for RBM operability. Plant personnel, after thoroughly reviewing this event, determined it was a condition that alone could have prevented the fulfillment of the safety function of the RBM to control (prevent) the release of radioactive material during an RWE.

This event was the result of a disparity between the core analysis design procedure assumptions and the Unit 1 and Unit 2 Technical Specifications requirements regarding RBM operability. As a corrective action, Operations procedure 34GO-OPS-065-0S, "Control Rod Movement," now requires the Unit 1 and Unit 2 RBMs to be operable whenever reactor power is above 27 percent rated thermal power. Proposed changes to the Unit 1 and Unit 2 Technical Specifications are being prepared for submittal to the NRC.

## PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

## DESCRIPTION OF EVENT

On 3/25/97 at 1307 EST, Unit 1 was in the Run mode at a power level of 2558 CMWT (100 percent rated thermal power) and Unit 2 was in the Refuel mode with the reactor pressure vessel head removed and the cavity flooded. At that time, plant personnel determined a core analysis procedure inconsistency with the plant's Technical Specifications involving the Rod Block Monitor (RBM, EIIS Code JD) met the reporting requirements of 10 CFR 50.72 (b)(2)(iii) and 50.73 (a)(2)(v). This disparity was reported to Southern Nuclear Operating Company personnel by General Electric personnel. General Electric reported that their core design procedures contained a step which allowed them to assume the RBM was operable and available to mitigate the consequences of a Rod Withdrawal Error (RWE) irrespective of the individual plant's Technical Specifications requirements for RBM operability.

Unit 1 and Unit 2 Technical Specifications Table 3.3.2.1-1 allows the RBM to be inoperable under certain conditions of reactor power and Minimum Critical Power Ratio, e.g., rated thermal power $\geq$ 90 percent and Minimum Critical Power Ratio $< 1.40$. However, General Electric core design procedures allowed them to assume the RBM is operable under these conditions. Consequently, under certain unlikely combinations of control rod position, core power, fuel bundle power, and Minimum Critical Power Ratio, the RBM may be needed to mitigate the consequences of an RWE transient, i.e., prevent violation of the fuel cladding design limit of one percent plastic strain. However, the Technical Specifications as currently written, would not require RBM to be operable.

The existing RWE transient analysis concludes no cladding damage will result because the cladding design limit of one percent plastic strain would not be exceeded. As mentioned previously, this analysis may rely upon the RBM to ensure the fuel cladding design limit is not violated when the Technical Specifications do not require it to be operable. Therefore, it is possible, albeit highly unlikely, an RWE may lead to some fuel cladding damage and result in the release of some radioactive materials from the fuel. Accordingly, plant personnel, after thoroughly reviewing this event, determined it was a condition that alone could have prevented the fulfillment of the safety function of the RBM to control (prevent) the release of radioactive material during an RWE event.

CAUSE OF EVENT

This event was the result of a disparity between General Electric's core design analysis procedure assumptions and the Unit 1 and Unit 2 Technical Specifications requirements regarding RBM operability. The plant's Technical Specifications allow the RBM to be inoperable under certain combinations of power level and Minimum Critical Power Ratio. However, General Electric's core design procedure contained a step which allowed them to assume the RBM was operable under these conditions. Therefore, a situation existed in which the analysis for an RWE assumed the RBM was available to mitigate the consequences of the transient, but the Technical Specifications did not require it to be operable as assumed in the analysis.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(v) because an analysis error was discovered which, had the error not been corrected, could have resulted in a challenge to one of the plant's principal safety barriers. The fuel vendor design analysis procedures allow the RBM to be assumed to be operable and available to mitigate the consequences of an RWE such that this transient does not cause the fuel design limit of one percent plastic strain to be exceeded for a given core design. However, the plant's Technical Specifications do not require the RBM to be operable under some conditions assumed in the RWE transient analysis. Therefore, under certain unlikely circumstances of control rod position, core power, fuel bundle power, and Minimum Critical Power Ratio with the RBM inoperable, an RWE may have resulted in fuel cladding damage otherwise not postulated to occur during the transient. The fuel cladding is the first barrier to the release of radioactive material.

A value of one percent plastic strain of the Zircaloy fuel cladding is defined as the limit below which fuel damage due to overstraining of the cladding is not expected to occur. The limit of one percent plastic strain is based upon General Electric data on the strain capability of irradiated Zircaloy cladding segments from fuel rods operated in several Boiling Water Reactors. A statistical distribution fit to the available data indicates the one percent plastic strain value to be approximately the 95 percent point in the total population. Therefore, it may be deduced that there is a 95 percent chance a fuel rod with plastic strain of one percent or less will not experience cladding failure.

The RBM is designed to prevent violation of the fuel cladding design limit of one percent plastic strain that may result from a single RWE event. The RBM is assumed to mitigate the consequences of an RWE event when the unit is operating at greater than or equal to 29 percent rated thermal power. Below this power level, the consequences of an RWE event will not exceed the one

percent plastic strain fuel cladding design limit and therefore the RBM is not required to be operable.

In this event, General Electric reported a core analysis procedure inconsistency with the plant's Technical Specifications, i.e., an analysis of an RWE event could be performed using assumptions of RBM operability that are not reflected in the plant's Technical Specifications. The Technical Specifications allow the RBM to be inoperable under certain combinations of reactor power and Minimum Critical Power Ratio. However, the RWE transient analysis may be performed assuming the RBM is operable under these conditions. Therefore, it is possible that under certain unlikely combinations of control rod position, core power, fuel bundle power, and Minimum Critical Power Ratio with the RBM inoperable, the fuel vendor's RWE analysis would predict no fuel cladding damage when in fact it could occur.

General Electric has analyzed the current Unit 1 fuel cycle using the projected, and, where possible, actual control rod patterns that existed during the cycle and assuming the RBM was operable consistent with the requirements of the present Unit 1 Technical Specifications. Their calculations showed that the fuel cladding design limit of one percent plastic strain would not have been exceeded for a postulated RWE event during the current Unit 1 cycle. General Electric also has reviewed the past four Unit 1 and five Unit 2 fuel cycles. In all cases, they determined sufficient margin existed such that the one percent plastic strain limit would not have been violated during a postulated RWE event. Therefore, an RWE transient would not have resulted in fuel cladding damage or the release of radioactive materials irrespective of the aforementioned core analysis procedure disparity with the plant's Technical Specifications.

Based upon the preceding analysis, it may be concluded this event had no adverse impact on safety. This analysis is applicable to all power levels at which the RWE transient is of interest.

## CORRECTIVE ACTIONS

Operations procedure 34GO-OPS-065-0S, "Control Rod Movement," effective 4/16/97, requires the Unit 1 and Unit 2 RBMs to be operable whenever reactor power is above 27 percent rated thermal power, regardless of the MCPR.

Proposed changes to the Unit 1 and Unit 2 Technical Specifications in which the RBMs will be required to be operable any time reactor power is greater than or equal to 29 percent rated thermal power are being prepared for submittal to the NRC.

General Electric has committed verbally to SNC to perform a review of their vendor/utility interface process that deals with the selection of important plant characteristics used in transient analyses for reload licensing applications. The purpose of this review is to ensure the current process provides assurance that the reload licensing analyses are consistent with the plant's licensing basis. As a test of this process, General Electric will review the Hatch Technical Specifications and their relationship with the reload licensing analysis process.

SNC's BWR Core Analysis group has obtained current copies of General Electric core design analysis and reload licensing analysis procedures and will perform an independent comparison of the procedures and Plant Hatch Technical Specifications requirements. The BWR Core Analysis group also will compare the General Electric procedures with transient and accident analyses assumptions contained in the Unit 1 and Unit 2 Final Safety Analysis Reports.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

No failed components caused or resulted from this event.

A previous similar event in which an analysis weakness resulted in an event which could have challenged a safety barrier needed to control the release of radioactive material was reported in Licensee Event Report 50-366/1996-001, dated 6/4/96. The previous similar event was the result of the methodology used by General Electric to calculate the generic Safety Limit Minimum Critical Power Ratio. The method used did not yield a conservative value for the safety limit for Hatch 2 Cycle 13. Corrective actions for that event would not have prevented this event because they addressed the limit affected by and causes for the previous event, both of which were specific to that event. The cause for this event, i.e., analysis assumptions not reflected in the Technical Specifications, could not have been detected or corrected by the corrective actions for the previous event.