NRC FORM 366
(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: [ ] (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

[0 1] [A L J M F 1] (2) [0 0 - 0 0 0 0 0 - 0 0] (3) [4] [1 1 1 1] (4) [ ] (5)
7 8 9 LICENSEE CODE 14 15 LICENSE NUMBER 25 26 LICENSE TYPE 30 57 CAT 58

CON'T
[0 1] REPORT SOURCE [L] (6) [0 5 0 0 0 3 4 8] (7) [0 3 3 0 7 9] (8) [1 0 2 9 7 9] (9)
7 8 60 61 DOCKET NUMBER 68 69 EVENT DATE 74 75 REPORT DATE 80

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

[0 2] Westinghouse notified Alabama Power Co. that a review of safety analysis methodology
[0 3] for the single dropped rod indicated a potential for that event to lead to calculated
[0 4] DNB ratios lower than reported to the NRC for the FNP class of plant. The impact of
[0 5] this inconsistency is believed to be minimal in that there are several mitigating
[0 6] effects not credited in the calculational method that significantly reduce the conse-
[0 7] quences of this transient (see attachment). Engineering review of this notification
[0 8] resulted in a determination on 3/30/79, that this inconsistency is potentially report-
7 8 9 able under Tech. Spec. 6.9.1.8(h). 80

[0 9] SYSTEM CODE [Z Z] (11) CAUSE CODE [X] (12) CAUSE SUBCODE [Z] (13) COMPONENT CODE [Z Z Z Z Z Z] (14) COMP. SUBCODE [Z] (15) VALVE SUBCODE [Z] (16)
7 8 9 10 11 12 13 18 19 20

(17) LER/RO REPORT NUMBER: EVENT YEAR [7 9] [—] SEQUENTIAL REPORT NO. [0 1 7] [/] OCCURRENCE CODE [0 1] REPORT TYPE [T] [—] REVISION NO. [3]
21 22 23 24 26 27 28 29 30 31 32

ACTION TAKEN [X] (18) FUTURE ACTION [X] (19) EFFECT ON PLANT [Z] (20) SHUTDOWN METHOD [Z] (21) HOURS [0 0 0 0] (22) ATTACHMENT SUBMITTED [Y] (23) NPRD-4 FORM SUB. [N] (24) PRIME COMP. SUPPLIER [Z] (25) COMPONENT MANUFACTURER [Z 9 9 9] (26)
33 34 35 36 37 40 41 42 43 44 47

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

[1 0] This potential inconsistency arose from two sources (a) the existing rod controller
[1 1] can potentially incorrectly measure the core average power level during certain rod
[1 2] drop events (b) deviations between rod control settings actually used in the field and
[1 3] those assumed at the time of the safety analysis. Changes to the Power Range N.I.
[1 4] negative and positive rate trip setpoints have been made. (See Attachment)
7 8 9 80

[1 5] FACILITY STATUS [H] (28) % POWER [0 0 0] (29) OTHER STATUS (30) [NA] METHOD OF DISCOVERY [D] (31) DISCOVERY DESCRIPTION (32) [Notification from NSSS vendor]
7 8 9 10 12 13 44 45 46 80

[1 6] ACTIVITY RELEASED [Z] (33) CONTENT OF RELEASE [Z] (34) AMOUNT OF ACTIVITY (35) [NA] LOCATION OF RELEASE (36) [NA]
7 8 9 10 11 44 45 80

[1 7] PERSONNEL EXPOSURES NUMBER [0 0 0] (37) TYPE [Z] (38) DESCRIPTION (39) [NA]
7 8 9 11 12 13 80

[1 8] PERSONNEL INJURIES NUMBER [0 0 0] (40) DESCRIPTION (41) [NA]
7 8 9 11 12 80

1269 281

[1 9] LOSS OF OR DAMAGE TO FACILITY TYPE [Z] (42) DESCRIPTION [Na]
7 8 9 10 80

[2 0] PUBLICITY ISSUED [N] (43) DESCRIPTION (45) [NA] NRC USE ONLY
7 8 9 10 68 69 80

7911050 301

NAME OF PREPARER W. G. Hairston, III PHONE (205) 899-5156

ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT
DOCKET NO. 50-348
ATTACHMENT TO LER 79-017/01T-3

Facility: Joseph M. Farley Unit 1

Report Date: 10/29/79

Event Date: 3/30/79

Identification of Event

Westinghouse notification was received of a safety analysis methodology inconsistency in the NRC reviewed single dropped rod analysis.

Conditions Prior to Event

The unit was in mode 6 at the time of notification.

Description of Event

Westinghouse notified Alabama Power Company that a review of safety analysis methodology for the single dropped rod indicated a potential for that event to lead to calculated DNB ratios lower than reported to the NRC for the FNP class of plant. Engineering evaluation of this notification resulted in a determination on 3/30/79 that this inconsistency is potentially reportable under Tech. Spec. 6.9.1.8(h).

Designation of Apparent Cause

This potential inconsistency arose from two sources:

a. The existing rod controller can potentially incorrectly measure the core average power level during certain rod drop events.

b. Deviations between rod control settings actually used in the field and those assumed at the time of the safety analysis.

Analysis of Event

The impact of this inconsistency is believed to be minimal in that there are several mitigating effects not credited in the calculational method that significantly reduce the consequences of this transient. Among these effects are:

1. To meet the F_q limits required for LOCA and reduce burnup shadowing the control rods during normal operation are typically inserted less than 5 to 10%. This corresponds to approximately 100 pcm of reactivity. The dropped rod that is assumed has a worth that is also typically 100 pcm. Thus, the rod controller, by withdrawing the control bank, can restore full power but generally cannot result in a power overshoot.

2. The dropped rod assumed is the most limiting rod in terms of the resulting increase in $F_{\Delta H}$. The majority of rods, if dropped, would result in much lower increases in $F_{\Delta H}$.

3. Our analysis assumes that the negative flux trip does not occur during this transient. However, the available plant data indicate that most single dropped rods result in a negative flux rate trip. In fact, these rods that provide the limiting $F_{\Delta H}$ values are also most likely to provide a reactor trip.

4. The analysis presented in the FSAR assumed the bounding conservative reactivity coefficients allowed by the Tech. Spec. The actual reactivity coefficients in the plant are significantly less limiting. The use of the actual moderator and doppler coefficients would reduce the power overshoot.

5. The FSAR analysis did not assume the operation of the overpower rod block because it is control grade equipment. This block is expected to be in operation and would terminate rod motion when the power increases to 103% of nominal. This effect greatly reduces the potential for an excessive power overshoot.

6. Improved safety analyses methods reviewed and approved by the NRC on the D. C. Cook Unit 2 application (use of statistical DNB and the WRB-1 correlation), demonstrate the existence of significant margins compared to the margins shown in the original safety analyses for affected plants. Recognition of this margin, as well as other conservative features of our overall safety analysis methodology, can provide high assurance that the single rod drop event does not, in fact, violate the accepted limiting DNB ratio.

7. The rod control system limits the amount of power overshoot during the rod drop transient. For those plants where this effect is significant, minor modifications to the settings will further reduce the magnitude of power overshoot. Recognizing the importance of the rod control system performance during this transient, it is prudent to provide means in the form of procedures to ensure that the rod control system performs as designed.

Effect on Plant

This occurrence had no effect on plant operation.

Corrective Action

In accordance with Westinghouse recommendations, the Power Range Nuclear Instrumentation negative rate trip setpoint has been changed from $\leq$ 5% of rated thermal power with a time constant $\geq$ 2 seconds to $\leq$ 3% of rated thermal power with a time constant of $\geq$ 1 second.

Also, the Power Range Nuclear Instrumentation positive rate trip setpoint has been changed from $\leq$ 5% of rated thermal power with a time constant $\geq$ 2 seconds to $\leq$ 5% of rated thermal power with a time constant $\geq$ 1 second.

This setpoint change will protect the plant against single rod drop accidents thus precluding possible DNBR violation subsequent to a rod drop. Plant procedures have been revised to reflect the NRC approved Technical Specification change regarding these setpoint changes.

Failure Data

None