NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT HATCH, UNIT 2	0 5 0 0 0 3 6 6	1 OF 0 7

TITLE (4): DEFICIENT PROCEDURE RESULTS IN INADEQUATE SURVEILLANCE RESULTS

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 NAMES	DOCKET NUMBER(S)
0 4	1 4	8 8	8 8	— 0 1 0 —	0 0	0 5	1 3	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 0

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)	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)	X 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
J. D. Heidt, Nuclear Licensing Manager - Hatch	AREA CODE 404 526-4530

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): YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On 04/14/88 at approximately 1445 CDT, a member of the Procedure Upgrade Program (PUP) determined that the response time testing procedure for the Average Power Range Monitor circuitry (APRM EIIS Code IG) did not produce data which could consistently demonstrate that the flow referenced, upscale Simulated Thermal Power Trip (STPT) response times were acceptable. Specifically, the procedure did not effectively exclude the capacitative (RC) time constant in the STPT circuit, as permitted by the Technical Specifications. Therefore, from the resulting data, it could not always be determined whether the response time acceptance criteria had been met, resulting in a condition prohibited by the plant's Technical Specifications.

The root cause is procedure inadequacy. The procedure did not properly exclude the simulated thermal power time constant from the measurement of the response time.

The corrective actions for this event included scheduling the revision or replacement of the deficient procedure and its satisfactory performance prior to the startup from the next Unit 2 refueling outage.

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A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(i), because a condition existed that was prohibited by the plant's Technical Specifications. Specifically, some of the surveillance requirements of Technical Specifications section 4.3.1.3 were not conclusively met.

B. UNIT(s) STATUS AT TIME OF EVENT

1. Power Level/Operating Mode

Unit 2 was in steady state operation at an approximate power level of 2436 MWt (approximately 100 percent of rated thermal power). The reactor mode switch was in the run position.

2. Inoperable Equipment

There was no inoperable equipment that contributed to this event.

C. DESCRIPTION OF EVENT

1. Event

On 4/14/88 at approximately 1445 CDT, a member of the Procedure Upgrade Program (PUP) was performing a review of procedure 57SV-C51-002-2S (APRM Time Response Test). At that time, the PUP member determined that the procedure's method of testing the Average Power Range Monitor circuitry (APRM EIIS Code IG) did not produce data which could consistently demonstrate that the flow referenced, upscale Simulated Thermal Power Trip (STPT) response times were acceptable. A Deficiency Card was generated, as required by the plant's administrative control procedures, to document the condition.

Specifically, Technical Specifications section 4.3.1.3 requires that the Reactor Protection System (RPS EIIS Code JC) response time of each reactor trip function listed in table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Procedure 57SV-C51-002-2S was intended to enable meeting the requirement of item 2.b. of table 3.3.1-2. The response time of the flow referenced STPT is required to be less than or equal to 0.09 seconds, not including the simulated thermal power time constant.

The STPT circuitry is made up of two key components, a thermal power simulator and a flow referenced trip unit. The thermal power simulator conditions the APRM neutron flux signal through a first order low pass filter that has a 6 second capacitative (RC) time constant. Thus, the fuel time constant is approximated which causes the neutron flux to lead the thermal power during power increase events.

Procedure 57SV-C51-002-2S provided time response acceptance criteria of less than or equal 0.09 seconds for the STPT circuit. However, it did not provide for excluding the simulated thermal power time constant portion of the circuitry for the STPT response time tests. Instead a note in the procedure stated that the measured response time would include the simulated thermal power time constant of 6 seconds.

With no further information on the nature of the time constant, the non-licensed Instrumentation and Control (I&C) personnel responsible for performing the procedure added an assumed maximum 6 second time delay to the 0.09 Technical Specifications limit. As a result, they implemented the procedure by finding any measured response time less than or equal to 6.09 seconds acceptable.

The procedural data packages resulting from performance of procedure 57SV-C51-002-2S (and its predecessor HNP-2-3198) during the years 1980-1988 were reviewed. Response times measured during the years 1980-1986 were all less than 6.09 seconds. However, it could not be conclusively determined what portion of the measured time is attributable to the simulated thermal power time constant circuit.

With an RC time constant, the actual measured response time is related to the applied test signal (input voltage). The data packages do not include information on the application of the input voltage. Therefore, for the years 1980-1986, it could not be determined conclusively whether the response time requirement of 0.09 seconds for the STPT, not including the simulated thermal power time constant, was met.

However, the procedure data packages for 1987 document response times which were all less than 0.09 seconds. The input voltages used in 1988 were sufficient for the complete STPT circuit to respond within 0.09 seconds. Thus, based on the 1988 data, the ability of the STPT to meet the response time acceptance criteria of the Technical Specifications was conservatively demonstrated.

Further investigation showed that the STPT had been tested by using the erroneous 6 second time delay assumption since the STPT was installed per Design Change Request 79-94 on 5/1/80. Therefore, unless the resulting response times were less than 0.09 seconds for the complete STPT circuit, as tested, the acceptability of the STPT response times per Technical Specifications requirements was not conclusively demonstrated during this time period.

2. Dates/Times

Date	Time (CDT)	Description
4/14/88	1445	PUP personnel determined that procedure 57SV-C51-002-2S did not correctly incorporate the testing requirements of the Technical Specifications regarding the APRM STPT response time. Therefore, the resulting procedure data packages for 1980-1986 cannot conclusively demonstrate acceptable response times. A Deficiency Card was generated to document the condition. The 1988 procedure data packages were found to conservatively demonstrate acceptable STPT response times.

3. Other Systems Affected

No safety systems, other than the APRM STPT portion of the RPS, were affected by this event. This system has no secondary functions.

4. Method of Discovery

This event was discovered as a part of the PUP. This is a long term program to upgrade all plant procedures. For surveillance procedures, the PUP includes a technical review to ensure that these procedures properly address all Technical Specifications requirements. Procedure 57SV-C51-002-2S had not yet been through the PUP.

5. Operator Actions

No operator actions were required in this event.

6. Auto/Manual Safety System Response

No manual or automatic safety systems actuations occurred, nor were any required to occur.

D. CAUSE OF EVENT

1. Immediate Cause

The immediate cause of this event is the same as the root cause.

2. Root/Intermediate Cause

The root cause of this event is deficient procedure 57SV-C51-002-2S. The procedure erroneously allowed for a 6 second time constant to be included in the STPT time response acceptance criteria. A study of the system has shown that the 6 second time delay was an incorrect assumption because a variable delay from 0 to 30 seconds is introduced by the simulated thermal power time constant circuit, dependent on actual test conditions.

Plant personnel who developed procedure HNP-2-3198 (APRM Response Time Test) Revision 0 wrote it such that the design's 6 second time constant circuit could be mistook for a 6 second time delay by the plant personnel implementing the procedure. Consequently, the simulated thermal power time constant was not effectively excluded from the STPT response time test. Procedure HNP-2-3198 was developed upon installation of the STPT in 1980, and the pertinent steps were later incorporated into procedure 57SV-C51-002-2S.

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PLANT HATCH, UNIT 2	0 5 0 0 0 3 6 6	8 8	0 1 0	0 0	0 6 OF 0 7

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

E. ANALYSIS OF EVENT

The APRM flow referenced STPT was installed in 1980 to reduce the number of spurious scrams occurring along the power-flow line without reducing the fuel safety margins for any accidents or abnormal operational transients for which the plant is licensed. The STPT accomplished this reduction in spurious scrams by replacing the flow referenced neutron flux trip. The STPT augments a fixed upscale neutron flux trip.

The spurious scrams were caused by neutron flux spikes due to momentary flow changes in the recirculation system flow or small pressure disturbances during turbine stop valve and turbine control valve testing. These small neutron flux spikes represented no danger to the fuel, because their duration was less than the fuel thermal time constant. Therefore, the fuel surface heat flux did not increase sufficiently to challenge the fuel cladding integrity safety limit.

The STPT receives indications of neutron flux from APRM channels and processes the signal through a time delay circuit. This circuit approximates the fuel dynamics to give an indication of the fuel surface heat flux and reactor power during both steady state and transient conditions. The signal is then referenced to core flow and input to a trip unit which provides the RPS trip function on simulated thermal power.

The FSAR analyses do not take credit for the APRM STPT to mitigate the consequences of any accident and generally do not take credit for it to mitigate transients. Pressurization transients typically establish the thermal margins to the fuel cladding integrity safety limit. These transients are assumed to result in scrams due to other scram signals, such as high reactor vessel dome pressure or high neutron flux.

Of the cold water injection transients, only the loss of feedwater heating (LFWH) transient takes credit for the STPT. The LFWH transient has not set Minimum Critical Power Ratio (MCPR) limits for Plant Hatch. In addition, it is a relatively slow transient (over a minute) so the effect of the fuel time constant and instrument response time is insignificant.

In this event, the procedural discrepancy resulted only in the response time of the STPT not being conclusively measured. The STPT setpoints and functional capabilities have been tested in accordance with Technical Specifications requirements under other procedures.

Therefore, since credit is not generally taken for the STPT in FSAR transient and accident analyses and instrument response time is not an important parameter in the LFWH transient, it is concluded that this event had no adverse impact on nuclear safety. The above analysis is applicable to all power levels.

F. CORRECTIVE ACTIONS

The corrective actions for this event included:

1. Scheduling revision or replacement of procedure 57SV-C51-002-2S (APRM Time Response Test) to support its performance prior to startup from the next Unit 2 refueling outage. The replacement procedure will meet the response time test requirement of Technical Specifications table 3.3.1-2, item 2.b, by bypassing the simulated thermal power time constant circuit, as permitted by Technical Specifications footnote to item 2.b.

G. ADDITIONAL INFORMATION

1. FAILED COMPONENT(s) IDENTIFICATION

There was no component failure experienced in this event.

2. PREVIOUS SIMILAR EVENTS

There are no previous similar events where the APRM system was not properly tested for response times.

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the southern electric system

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May 13, 1988

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

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
DEFICIENT PROCEDURE RESULTS
IN INADEQUATE SURVEILLANCE RESULTS

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a condition that was prohibited by the plant's Technical Specifications. The event occurred at Plant Hatch - Unit 2.

Sincerely,

R. P. McDonald
Executive Vice President,
Nuclear Operations

CLT/ct

Enclosure: LER 50-366/1988-010

c: (see next page)

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Georgia Power

c: Georgia Power Company
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Mr. L. T. Gucwa, Manager Nuclear Safety and Licensing
GO-NORMS

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Mr. L. P. Crocker, Licensing Project Manager - Hatch

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