UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

50-219

JAN 0 3 1983

MEMORANDUM FOR: Richard W. Starostecki, Director
Division of Project and Resident Programs, Region I

FROM: Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

SUBJECT: EVALUATION OF OYSTER CREEK LERs COVERING THE PERIOD
DECEMBER 1, 1981 TO NOVEMBER 30, 1982

In support of the upcoming SALP review of the Jersey Central Power and Light Company, in regard to their performance as licensee of the Oyster Creek Nuclear Generating Station, AEOD has assessed the Licensee Event Reports submitted under Docket No. 50-219 during the subject period. Our perspective is indicative of a a knowledgeable BWR system safety engineer, who is not, however, intimately familiar with the detailed site-specific equipment arrangements and operations. Our review focused on the technical accuracy, completeness and intelligibility of the LERs. Our review covered a majority of the LERs submitted.

In general the submittals were uniformily outstanding on the above points. The LERs typically contained very good descriptions of the events as well as excellent explanations of the consequence of the event on both the effected system performance level and the overall plant safety level. Furthermore, cause descriptions were typically very well documented, often providing both root cause information and symptomatic (or secondary) failure cause information. Finally, corrective action generally were considered to be appropriate and well described.

If you or your staff have any questions regarding this matter, please contact Stuart Rubin at 492-4436.

Karl V. Seyfrit

Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

cc: J. Lombardo, NRR
J. Thomas, RI

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CF/ADOCK/50-219

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 12 1983

MEMORANDUM FOR: Richard Lewis, Director
Division of Project and Resident Programs
Region II

FROM: Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
for Operational Data

SUBJECT: EVALUATION OF HATCH UNITS 1 AND 2 LERs FOR THE PERIOD
JULY 1, 1981 to OCTOBER 31, 1982

The office for Analysis and Evaluation of Operational Data has assessed the Licensee Event Reports submitted under Docket Nos. 50-321 and 50-366 during the subject period. This has been done in support of the upcoming SALP review of the Georgia Power Company, with regard to their performance as licensee of the Edwin I. Hatch Nuclear Power Plants. Our perspective would be indicative of that of a BWR system safety engineer, who although knowledgeable, is not intimately familiar with the detailed site-specific equipment arrangements and operations. Our review focused on the technical accuracy, completenes and intelligibility of the LERs. Additionally, the LERs were screened and sorted in an attempt to call out qualitative trends or patterns which could be interpreted as suggestive of licensee performance needing improvement. Our review covered a majority of the LERs submitted during the assessment period.

In general the LER submittals were acceptable with respect to the short alphanumeric fields on the LER form. However, they were usually minimally adequate with regard to the completeness of the narrative sections. There appeared to be a general tendency of providing no more descriptive information than the available space allowed on the form. When a supplemental sheet was provided, it frequently simply repeated the information provided on the LER form. The information, to the extent it was provided, appeared to be technically accurate and understandable however.

Our screening of the LERs for trend and patterns provided many cases which appeared to be indicative of management weakness of one sort or another. The sheer quantity of LERs submitted (almost 200 for each unit) in and of itself suggests this to be the case. More importantly, however, we considered an unusually large number of these to be "management deficiency related". These included numerous cases where: surveillance tests were performed either late or incorrectly; operating personnel actions or activities were incorrect; trade or technician workmanship was deficient; procedures were inadequate, improper or lacking altogether. The enclosures provide further detail of our assessment for Hatch Units 1 and 2.

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If you have any questions regarding this matter please contact either myself or Stuart Rubin of my staff.

Karl V. Seyfrit

Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

Enclosure:
As stated

cc: M. Fairtile, NRR
C. Michelson, AEOD

Enclosure 1

Hatch 1 LER Assessment

1. Incomplete LERs

The attached licensee event report (82-030) provides an example of a poorly written LER on several points as follows.

a) The author does not identify the name (i.e. purpose) of the valve, which did not operate adequetly. Without a P&ID for the system it is not immediately clear which valve is being described. This makes it difficult to analyze, evaluate and encode the information provided. The purpose or function of the component should be briefly described.

b) The effect (i.e. consequences) on the system and/or its function is not explained. GPC's usual description of the probable consequences is that "Public health and safety were not affected by this incident". It is important that the licensee describe the potential consequences at the system level (i.e. loss of function?, degraded performance?) and at the overall plant safety level (i.e. backup or redundant system/component availabilities).

c) The cause description does not provide the root or underlying cause for the event. That is, although it is stated that the mechanical overload switch for closure was found set at "minimum operator requirements", it does not explain why this occurred. Was the cause due to human error, inadequate procedures, a faulty switch setpoint adjustment mechanism or some other reason? There is no indication of any attempt having been made to investigate the root cause. If a limited underlying cause investigation was in fact performed, the results are not described. That is, there is no statement: "The results of an investigation could not establish the precise cause". Accordingly, it can only be concluded that corrective actions beyond resetting the setpoint were either not developed or not described.

d) Further detail is not provided even though a supplemental sheet containing a narrative discussion has been attached to the LER. Supplemental information is important as it enables the reader to more fully understand and better evaluate the event. Pertinent supplemental information should be furnished where it is know by the licensee. This is frequently not done by CPL.

2. Failure to Update LERs

LER 82-028 dated May 7, 1982 provides an example of an apparently un-updated LER.

The subject LER states in part that "an updated LER will be submitted to the NRC before startup." By the end of the assessment period no updated LER could be found from our sources of information.

3. LERs which Suggest Management Weaknesses

The cause for a significant fraction of the reported events reviewed could be considered as traceable to management oversights or omissions of one sort or another. The following lists such LERs by category.

a) Late Surveillance Test:

81-098
81-111
81-116
81-119
82-015
82-027
82-050
82-059

b) Incorrect Interpretation of T.S. Requirements:

81-130

c) Incorrect Testing/Personnel Actions or Activities:

81-056
81-062
81-066
81-072
81-073
82-019
82-035
82-046
82-049
82-053
82-055
82-064
82-066

d) Poor Electrical/Mechanical Workmanship/Design Activities:

81-049
81-050
81-071
81-105
81-122
81-133
81-137
81-140
82-022

e) Improper or Indequate Procedures/Control:

81-055
81-057
81-124
81-134
82-034
82-038
82-052

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Georgia Power

Edwin I. Hatch Nuclear Plant

PM-82-428
May 13, 1982

82 MAY 17 P 3: 37

USNRC REGION II
ATLANTA, GEORGIA

PLANT E. I. HATCH
Licensee Event Report
Docket No. 50-321

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
Suite 3100
101 Marietta Street
Atlanta, Georgia

ATTENTION: Mr. James P. O'Reilly

Pursuant to Section 6.9.1.9.b of Hatch Unit I Technical Specifications, please find attached Reportable Occurrence Report No. 50-321/1982-030.

for H. C. Nix
Plant Manager

HCN/RTN/mla

xc: J. H. Miller, Jr.
R. J. Kelly
G. F. Head
J. T. Beckham, Jr.
H. L. Sumner
R. D. Baker
Control Room
File

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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 | G A E I H 1 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | | (5)

LICENSEE CODE 9–14 · LICENSE NUMBER 15–25 · LICENSE TYPE 26–30 · CAT 57–58

CON'T

0 1 | REPORT SOURCE L (6) | 0 5 0 0 0 3 2 1 (7) | 0 4 2 1 8 2 (8) | 0 5 1 3 8 2 (9)

DOCKET NUMBER 61–68 · EVENT DATE 69–74 · REPORT DATE 75–80

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On April 21, 1982, with unit I at steady state operation, while performing

0 3 HNP-1-3302, HPCI MOV Operability, the 1E41-F011 opened per procedure time

0 4 requirement, but failed to fully close with the control switch. Public

0 5 health and safety was not affected by this incident. This is a non-re-

0 6 petitive event.

0 7

0 8

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	S F (11)	E (12)	A (13)	V A L V E X (14)	E (15)	D (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 2	—	0 3 0	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
E (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	A (25)	L 2 0 0 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The cause of the valve failing to fully close was due to the closing

1 1 mechanical overload switch being set at the minimum operator requirements.

1 2 The switch was reset to the operator recommended setting and HNP-1-3302

1 3 was performed satisfactorily.

1 4

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	E (28)	0 9 9 (29)	NA	B (31)	Surveillance Test

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

PUBLICITY

2 0 ~~8205210248~~ 820513 PDR ADOCK 05000321 PDR — NA

NRC USE ONLY

LER No.: 50-321/1982-030
Licensee: Georgia Power Company
Facility: Edwin I. Hatch
Docket No.: 50-321

Narrative Report
for LER 50-321/1982-030.

On April 21, 1982, with Unit I at steady state power operation, while performing HNP-1-3302 (HPCI MOV Operability), the 1E41-F011 opened per procedure time requirements but failed to fully close with the control switch. The 1E41-F011 is a redundant shut off valve to the Condensate Storage Tank and not in the injection flow path. Public health and safety was not affected by this incident. This is a non-repetitive.

Upon investigating the cause of the failure, it was found that the close mechanical overload switch for the valve operator was set at the minimum setting. The switch was reset to the operator recommended setting and the valve operated correctly.

CONFIRMATION STATEMENT

For Document R.O. 1-82-30
(Description of Document)

I have checked the statements made in this document and, to the best of my knowledge, the statements made in this response are accurate.

H. E. [signature]
(Signature)

5-1-82
(Date)

REGULATORY COMPLIANCE REVIEW
DOCUMENT # LER 1-82-30
CHANGES NEEDED (✓) Yes ( ) No IF YES, SEE COMMENTS
COMMENTS: corrections in red
REVIEWED BY: TKS
DATE: 5/7/82

Enclosure 2

Hatch 2 LER Assessment

1. Incomplete LERs

The attached licensee event report Nos. 82-81, 82-91 and 82-101, submitted by GPC for Hatch Unit 2 are examples of poorly prepared LERs for several reasons. (The circumstances described in these LERs all relate to the same event which occurred on August 25, 1982. As background, a brief, but more integrated narrative description of this occurrence, as provided informally by GPC, is contained in an attachment to this enclosure. Additionally, a second attachment is provided which documents a preliminary outline of the series of events on that date as communicated by the Hatch 2 resident inspector shortly after the event.)

a) The three LERs do not cross reference each other anywhere in the text. Unless one is familiar with the event through some other means, it would be unlikely that the reader (who at best would be reading them at widely separated times) would associate the LERs with one another. The three LERs were narrowly written, camouflaging their relationship within the event.

b) The sum of the three LERs do not provide, nor even suggest, the full picture of plant systems interactions involved in the August 25, 1982 event. For example none of the occurrences in the secondary containment mentioned in the attachments are discussed in the LERs.

c) The summary of events provided by GPC in the attachments states that RCIC tripped at 0510 on 8/25/82, was rolled back up, but isolated on high turbine exhaust diaphram pressure. This equipment failure is not noted in any of the above LERs. Although it is not certain, LER 82-100 (attached) may be accouning for this failure. However, the event date in 82-100 is stated to be 8/28/82. Furthermore, from the failure description provided in the LER the reader would not be led to believe that the RCIC failure was a failure on demand. A failure on demand would be more serious than if the failure were discovered as part of routine serveillance tests which LER 82-100 suggests. Thus either no LER was written for the 8/25/82 RCIC failure or the LER provided (82-100) has hidden the importance of the failure.

2. Failure to Update LERs

The following table lists LERs in which it was stated (or suggested) that an updated report would be provided at a later date. The table also notes whether or not an updated report was received within the review period.

LER Number	Update Received
81-101	No
31-102	No
81-104	Yes
81-115	No
81-121	No

81 127	No
82-018	No
82-022	Yes
82-023	No
82-043	No
82-068	No

3. LERs which Suggest Management Weakneses

The cause for a significant fraction of the reported events reviewed could be considered as traceable to management oversights or omissions of one sort or another. The following tabulates by category such LERs.

a) Late Activities:

81-062
81-069
81-124
82-005
82-027

b) Incorrect Interpretation of T.S. Requirements:

82-051

c) Incorrect Testing/Personnel Actions or Activities:

81-055
81-113
81-128
82-003
82-030
82-036
82-037
82-038
82-041
82-048
82-052
82-053
82-071
82-077
82-080

d) Poor Electrical/Mechanical Workmanship/Design Activities:

81-060
81-071
81-077
81-106
81-118
81-125
82-019
82-042
82-054
82-083

e) Improper or Inadequate Procedures:

81-067
81-099
81-124
81-132
82-010
82-016
82-028
82-035
82-055
82-066
82-070
82-074

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Georgia Power

Edwin I. Hatch Nuclear Plant

82 SEP 27 A10 : 17

PM-82-939
September 21, 1982

PLANT E. I. HATCH
Licensee Event Report
Docket No. 50-366

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

ATTENTION: Mr. James P. O'Reilly

Pursuant to section 6.9.1.9.b of Hatch Unit Two Technical Specifications, please find attached Reportable Occurrence Report No. 50-366/1982-081.

H. C. Nix
Plant Manager

HCN/SBT/abb

xc: R. J. Kelly
G. F. Head
J. T. Beckham, Jr.
P. D. Rice
K. M. Gillespie
H. L. Sumner
S. B. Tipps
R. D. Baker
Control Room
Document Control

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LICENSEE EVENT REPORT

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

0 1 | C A E I H 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T 0 1 REPORT SOURCE L (6) | 0 5 0 0 0 3 6 6 (7) | 0 8 2 5 8 2 (8) | 0 9 2 1 8 2 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 With the unit at hot shutdown following a reactor scram, the Torus high
0 3 water level alarm was activated. The highest indicator showed a water
0 4 level of approximately 12 feet 7 inches. T.S.3.6.2.1.a requires a water
0 5 level between 12 feet 2 inches and 12 feet 6 inches. As per action item
0 6 a. of this Tech. Spec., the unit was at cold shutdown within 24 hours of
0 7 the event. The health and safety of the public were not affected. This
0 8 event is not repetitive.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	S H (11)	E (12)	B (13)	V A L V E X (14)	E (15)	D (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 2	—	0 8 1	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	A (25)	R 3 4 0 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The cause of this event was a reactor scram and group I isolation that
1 1 occurred as a result of a MSIV failing and going closed. Steam relief
1 2 valves opened to control reactor pressure discharged to the Torus. The
1 3 Torus water level was returned to within TS limits before the unit was
1 4 taken from cold shutdown. The MSIV was repaired and returned to service

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	0 0 0 (29)	NA	A (31)	Alarm

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

PUBLICITY ISSUED DESCRIPTION (45) NA

~~8209300162~~ 820921
PDR ADOCK 05000366
S PDR

NRC USE ONLY

LER No.: 50-366/1982-081
Licensee: Georgia Power Company
Facility: Edwin I. Hatch
Docket #: 50-366

Narrative Report
for LER 50-366/1982-081

On August 25, 1982, with the unit at hot shutdown following a reactor scram, the suppression chamber (Torus) high water level alarm was received in the control room. One indicator showed a water level of approximately 12 feet, 7 inches. Tech. Specs. section 3.6.2.1.a. states that the suppression chamber (Torus) shall be operable with a water volume equivalent to a water level between 12 feet 2 inches and 12 feet 6 inches. The limiting condition for operation (LCO) of Tech. Specs. section 3.6.2.1., Action item a. was complied with since the unit was already at hot shutdown and was at cold shutdown within 24 hours of this event. The health and safety of the public were not affected. This event is non-repetitive.

The cause of this event was a reactor scram and group I isolation that occurred as a result of a Main Steam Isolation Valve (MSIV) failing and going closed. After the scram, the High Pressure Coolant Injection (HPCI) system and the Reactor Core Isolation Cooling (RCIC) system started to help control reactor pressure and maintain reactor water level. Steam Relief Valves "A" and "D" (opened to relieve reactor pressure) discharged to the Torus and caused the Torus water level to rise above the Tech. Specs. limit.

The Torus water level was returned to Tech. Specs. limits before the unit was taken from cold shutdown. The MSIV whose failure initiated this event was repaired and returned to service.

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912 537-9444

Georgia Power

Edwin I. Hatch Nuclear Plant

September 23, 1982
PM-82-947

'82 SEP 30 A 9:29

PLANT E. I. HATCH
Licensee Event Report
Docket No. 50-366

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II, Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

ATTENTION: Mr. James P. O'Reilly

Pursuant to Section 6.9.1.9.b of Hatch Unit Two Technical Specifications, please find attached Reportable Occurrence Report No. 50-366/1982-091.

H. C. Nix
Plant Manager

HCN/SBT/amh

xc: R. J. Kelly
G. F. Head
J. T. Beckham, Jr.
P. D. Rice
K. M. Gillespie
H. L. Sumner
S. B. Tipps
R. D. Baker
Control Room
Document Control

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PDR ADOCK 05000366

LICENSEE EVENT REPORT

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

LICENSEE CODE (2)	G A E I H 2
LICENSE NUMBER (3)	0 0 - 0 0 0 0 0 - 0 0
LICENSE TYPE (4)	4 1 1 1 1
CAT (5)	
REPORT SOURCE (6)	L
DOCKET NUMBER (7)	0 5 0 0 0 3 6 6
EVENT DATE (8)	0 8 2 5 8 2
REPORT DATE (9)	0 9 2 3 8 2

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On 8/25/82, with Unit 2 in hot shutdown and a scram recovery in progress

0 3 the "H" ADS safety relief valve failed to open manually (DR 2-82-219)

0 4 and the drywell pressure (DR 2-82-220) and temperature (DR 2-82-221)

0 5 exceeded Tech. Specs. requirements of .75 psig and 135°F. High drywell

0 6 pressure prevented the suppression chamber/drywell vacuum breakers from

0 7 opening (DR 2-82-222). The health and safety of the public were not

0 8 affected by this non-repetitive event.

SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP. SUBCODE (15)	VALVE SUBCODE (16)
S A	X	X	Z Z Z Z Z Z	Z	Z

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 2	—	0 9 1	/	0 3	L	—	0

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
X	Z	Z	Z	0 0 0 0	Y	N	Z	Z 9 9 9

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The cause of the event is unknown. It is believed that the "A" SRV tail-

1 1 pipe vacuum breaker remained open when the "A" SRV was actuated for a

1 2 second time. This would allow a steam release to the drywell. The dry-

1 3 well pressure and temperature were returned to allowable levels. All SRV

1 4 vacuum breakers were inspected and were found to be operable.

FACILITY STATUS (28)	% POWER (29)	OTHER STATUS (30)	METHOD OF DISCOVERY (31)	DISCOVERY DESCRIPTION (32)
X	0 0 0	Scram Recovery	A	Operator Observation

ACTIVITY RELEASED (33)	CONTENT OF RELEASE (34)	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
Z	Z	NA	NA

PERSONNEL EXPOSURES

NUMBER (37)	TYPE (38)	DESCRIPTION (39)
0 0 0	Z	NA

PERSONNEL INJURIES

NUMBER (40)	DESCRIPTION (41)
0 0 0	NA

LOSS OF OR DAMAGE TO FACILITY (43)

TYPE (42)	DESCRIPTION
Z	NA

PUBLICITY

ISSUED (44)	DESCRIPTION (45)	NRC USE ONLY
N	~~8210050084~~ 820923 PDR ADOCK 05000366 S PDR	

NAME OF PREPARER S. B. Tipps PHONE: (912) 367-7851

GPO 917-928

LER No: 50-366/1982-091
Licensee: Georgia Power Company
Facility: Edwin I. Hatch
Docket #: 50-366

Narrative Report
for LER 50-366/1982-091

On 8/25/82, with Unit 2 in hot shutdown and a scram recovery in progress, the "H" ADS safety relief valve failed to open manually and the drywell pressure and temperature exceeded Tech. Specs. requirements. T.S. 3.6.1.6 requires that drywell pressure be maintained less than .75 psig; however, drywell pressure reached 2.7 psig. T.S. 3.6.1.7 requires that average drywell air temperature be maintained less than 135°F. The high drywell pressure also prevented the suppression chamber/drywell vacuum breakers from opening during the "SUPPRESSION CHAMBER TO DRYWELL VACUUM BREAKER SYSTEM OPERABILITY" procedure. T.S. 4.6.4.1.a. requires that the suppression chamber/drywell vacuum breakers be proven operable within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves. The plant was placed in cold shutdown within the 24 hours as required by T.S. 3.6.4.1, Action b. The health and safety of the public were not affected by this non-repetitive event.

The cause of the "H" valve failure has been attributed to component failure. The manual control switch for the failed valve was found to have worn parts. The faulty switch was replaced. It is believed that the "A" SRV tailpipe vacuum breaker failed to shut when the "A" SRV was actuated for a second time. This would allow a steam release to the drywell. The pressure differential between the suppression chamber and the drywell was equalized. Cooling via the drywell chillers was restored. Subsequently, the drywell pressure and temperature returned to allowable levels. All SRV tailpipe vacuum breakers were inspected and found to be operable. The suppression chamber to drywell vacuum breaker was satisfactorily functionally tested per the "SUPPRESSION CHAMBER TO DRYWELL VACUUM BREAKER SYSTEM OPERABILITY" procedure.

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Georgia Power

Edwin I. Hatch Nuclear Plant

PM-82-940
September 28, 1982

PLANT E. I. HATCH
Licensee Event Report
Docket No.: 50-366

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

ATTENTION: Mr. James P. O'Reilly

Pursuant to section 6.9.1.9.b of Hatch Unit Two Technical Specifications, please find attached Reportable Occurrence Report No. 50-366/1982-101.

H. C. Nix
Plant Manager

HCN/SBT/abb

xc: R. J. Kelly
G. F. Head
J. T. Beckham, Jr.
P. D. Rice
K. M. Gillespie
H. L. Sumner
S. B. Tipps
R. D. Baker
Control Room
Document Control

LICENSEE EVENT REPORT

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

LICENSEE CODE (2)	G A E I H 2
LICENSE NUMBER (3)	0 0 - 0 0 0 0 0 - 0 0
LICENSE TYPE (4)	4 1 1 1 1
CAT (5)	
CON'T REPORT SOURCE (6)	L
DOCKET NUMBER (7)	0 5 0 0 0 3 6 6
EVENT DATE (8)	0 8 2 9 8 2
REPORT DATE (9)	0 9 2 8 8 2

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

With the unit at 1336 MWt during a normal power increase following a startup, the inboard Main Steam Isolation Valve (MSIV) in loop C failed closed. This event is contrary to Tech. Specs. 3.6.3, Table 3.6.3-1, item A.1 and 3.4.7. Continued operation is permitted under the limiting condition for operation of ACTION a.2. for both Tech. Specs. 3.6.3 and 3.4.7. The health and safety of the public were not affected by this non-repetitive event.

SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP. SUBCODE (15)	VALVE SUBCODE (16)
C D	E	B	V A L V E X	E	D

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 2	—	1 0 1	/	0 3	L	—	0

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
A	Z	A	B	0 1 0 6	Y	N	A	R 3 4 4

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

Examination of the removed parts by the manufacturer has determined the cause to be improper stem to disk thread engagement. The entire disk and stem assembly was replaced in both "C" loop MSIV's. The valves have been satisfactorily tested and returned to service.

FACILITY STATUS (28)	% POWER (29)	OTHER STATUS (30)	METHOD OF DISCOVERY (31)	DISCOVERY DESCRIPTION (32)
C	0 5 5	NA	A	Operational Event

ACTIVITY RELEASED (33)	CONTENT OF RELEASE (34)	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
Z	Z	NA	NA

PERSONNEL EXPOSURES

NUMBER (37)	TYPE (38)	DESCRIPTION (39)
0 0 0	Z	NA

PERSONNEL INJURIES

NUMBER (40)	DESCRIPTION (41)
0 0 0	NA

LOSS OF OR DAMAGE TO FACILITY (43)

TYPE (42)	DESCRIPTION
Z	NA

PUBLICITY

ISSUED (44)	DESCRIPTION (45)
N	NA

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NRC USE ONLY

LER No.: 50-366/1982-101
Licensee: Georgia Power Company
Facility: Edwin I. Hatch
Docket #: 50-366

Narrative Report
for LER 50-366/1982-101

On August 29, 1982, with unit 2 at 1336 MWt, during a normal power increase following a startup, the "C" inboard Main Steam Isolation Valve (MSIV) failed closed. This event is contrary to Tech. Specs. section 3.6.3, Table 3.6.3-1, item A.1. and Tech. Specs. section 3.4.7. Continued operation is permitted under the limiting condition for operation of ACTION a.2. for both Tech. Specs.. Reactor power was limited to 75% power to comply with steam flow limits for 3 steam line operation. The health and safety of the public were not affected by this non-repetitive event.

The cause of this event was component failure. Examination of the removed parts by the manufacturer has determined the cause to be improper stem to disk thread engagement. The entire disk and stem assembly was replaced in both the inboard and outboard "C" MSIV's. The valves have been satisfactorily tested and returned to service.

A generic review is being made by the manufacturer to determine if there are inherent problems associated with this failure. Unit one's MSIV's have a different manufacturer and are thus not affected by this event.

Georgia Power Company
Post Office Box 439
Baxley, Georgia 31513
Telephone 912 367-7781
912 537-9444

Georgia Power

Edwin I. Hatch Nuclear Plant

PM-82-945
September 23, 1982

PLANT E. I. HATCH
Licensee Event Report
Docket No: 50-366

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

ATTENTION: Mr. James P. O'Reilly

Pursuant to Section 6.9.1.9.b of Hatch Unit Two Technical Specifications, please find attached Reportable Occurrence Report No. 50-366/1982-100.

H. C. Nix
Plant Manager

HCN/SBT/amh

xc: R. J. Kelly
G. F. Head
J. T. Beckham, Jr.
P. D. Rice
K. M. Gillespie
H. L. Sumner
S. B. Tipps
R. D. Baker
Control Room
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PDR

LICENSEE EVENT REPORT

(PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

LICENSEE CODE: G A E I H 2 — LICENSE NUMBER: 0 0 - 0 0 0 0 0 - 0 0 — LICENSE TYPE: 4 1 1 1 1

REPORT SOURCE: L — DOCKET NUMBER: 0 5 0 0 0 3 6 6 — EVENT DATE: 0 8 2 8 8 2 — REPORT DATE: 0 9 2 3 8 2

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

While performing"RCIC TURBINE EXHAUST DIAPHRAGM PRESSURE INST. FT&C" monthly procedure(required by Item 5c of Tech. Specs. Table 3.3.2-1), two inoperative switches were discovered. Per action 26 for Item 5c of Tech. Specs. Table 3.3.2-1,Valves 2E51-F007&F008 were closed and the RCIC System was declared inoperable. The unit was put in a 14-day limiting condition for operation (LCO) per Tech. Specs.section 3.7.3 ACTION a. Public health and safety were not affected by this non-repetitive event.

SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP SUBCODE	VALVE SUBCODE
C E	E	D	I N S T R U	E	Z

LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 2	—	1 0 0	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
C	Z	Z	Z	0 0 0 0	Y	N	A	B 0 6 9

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

The cause of the event was component failure of the switches which had sustained water damage and had become corroded. The switches were replaced and successfully tested by performing "RCIC TURBINE EXHAUST DIAPHRAGM PRESSURE INST. FT&C" procedure. The RCIC system was returned to service August 29, 1982.

FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
C	0 1 0	NA	B	Routine Test/Inspection

ACTIVITY CONTENT RELEASED	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
Z	Z	NA	NA

PERSONNEL EXPOSURES: 0 0 0 — Z — NA

PERSONNEL INJURIES: 0 0 0 — NA

LOSS OF OR DAMAGE TO FACILITY: Z — NA

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LER No: 50-366/1982-100
Licensee: Georgia Power Company
Facility: Edwin I. Hatch
Docket #: 50-366

Narrative Report
for LER 50-366/1982-100

On August 28, 1982, with Unit 2 in run at 250 MWt., two inoperative switches were discovered during the performance of the regular monthly surveillance (RCIC TURBINE EXHAUST DIAPHRAGM PRESSURE INST. FT&C procedure) requir-d by Tech. Specs. table 4.3.2-1(5c). Valves 2E51-F007 and F008 were closed to isolate the RCIC system and make it inoperable per Tech. Specs. Table 3.3.2-1(5C) Action 26. The unit was put in a 14-day Limiting Condition for Operation (LCO) per Tech. Specs. Seciton 3.7.3 Action a. Public health and safety were not affected by this non-repetitive event.

The cause of the event was component failure of the switches which had sustained water damage and had become corroded. The switches were replaced and successfully tested by performing the "RCIC TURBINE EXHAUST DIAPHRAGM PRESSURE INST. FT&C" procedure. The RCIC system was returned to service August 29, 1982.

SUMMARY OF EVENTS SURROUNDING HATCH UNIT 2 SCRAM ON 8/25/82

I. Events Leading to Scram

At 0417, on August 25, 1982, while operating at rated conditions, a reactor scram and Group 1 isolation occurred. The alarm typer only showed a half group 1 isolation due to main steam line high flow but it is believed by all shift personnel that the process computer for some reason didn't show the other half group 1 isolation.

II. Recovery

Upon the MSIV isolation, reactor pressure increased to approximately 1090 psig per a narrow range pressure instrument and approximately 1095 psig per process computer. SRV D auto-opened to relieve pressure to approximately 900 psig with the manual operation of SRV A assisting the depressurization.

Reactor water level dropped to approximately -40" on post accident monitoring system A and -45" on post accident monitoring system B as soon as the MSIVs closed but was quickly restored as the SRVs operation reduced reactor pressure. MSR supplied steam to the RFPT for a short time, bringing water level up to the high level trip setpoint for HPCI and RCIC. HPCI and RCIC auto started but didn't inject prior to level reaching the hi level trip setpoint.

Upon resetting the group 1 isolation and using RCIC to control level, equalization around the MSIVs was started at 0420. At 0429 a reactor lo water level alarm was received. HPCI was manually initiated to restore reactor water level. Level and pressure stabilized at approximately 32" and 990 psig at 0432.

SRV "A" was manually opened again at 0449 to reduce reactor pressure and help in equalizing around the MSIVs. All 4 MSIVs (inboard and outboard) were opened at 0450.

At 0451, a DW high pressure scram occurred with all systems responding as designed. Upon looking at the charts, it appears that drywell pressure and torus-to-drywell dp both increased very quickly. It is possible that the operation of the "A" SRV had caused a failure of the "A" SRV discharge line vacuum breaker causing the rapid increase in drywell pressure. At the same time, the drywell chiller unit tripped and wouldn't restart.

RCIC tripped at 0510 and was reset, rolled back up and isolated on high turbine exhaust diaphragm pressure. RFPT 2A was put on feeding the reactor vessel at 0515.

The DW hi pressure override switches were used to vent the drywell per HNP-2-1906 at 0515. Operation had to bypass PCIS low RPV level override instead of PCIS hi drywell pressure override to open 2T48-AOV-F332A,B and F334A,B.

The RCIC deluge system was initiated when Health Physics called at 0525 reporting smoke in RCIC diagonal. It is speculated that the "smoke" was in fact steam resulting from the scram discharge header exhausting into the CRW system. Some steam is thought to have escaped out of the CRW system into the RCIC diagonal.

The high DW pressure cleared at 0740 and the scram was reset followed by reducing the unit to the cold shutdown condition.

III. Resolutions (Investigation Concluding 9-15-82)

During the MSIV outage on 9-12-82, to determine the cause of the isolation of MSL "C", maintenance personnel found the poppet off of the "C" inboard MSIV. With the poppet off, the valve disc was free to drop off the shaft. Steam flow would then push the valve closed, causing a pressure rise and ensuant flux rise as voids were compressed. This is believed to have been the cause of the high flux scram. The MSIV isolation in turn resulted from the initial reactor water level decrease after the reactor scram. The drywell chillers were unable to restart due to load shedding logic with the drywell high pressure signal present.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 29 1982

AEOD/E240

MEMORANDUM FOR: Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

FROM: Stuart D. Rubin, Lead Engineer
Reactor Systems 4
Reactor Operations Analysis Branch

SUBJECT: PRELIMINARY ACCOUNT OF EVENTS ASSOCIATED WITH A REACTOR TRIP AT HATCH UNIT 2 ON AUGUST 25, 1982

REFERENCES:
1. Daily Report Item for Hatch 2 on August 25, 1982.
2. Telecon with Hatch 2 Resident.

Event Description

At about 4:15 AM EST on the subject date, the Hatch Unit 2 reactor tripped from approximately 99% power, as a result of an overpower condition on the Average Power Range Monitor (APRM - Hi). The high APRM flux was due to a transient reactor pressure increase, caused by the spurious closure of one of the two isolation valves on the "C" main steam line. The reactor trip, coupled with the transient reactor pressure increase resulted in a core void collapse which was sufficient to shrink vessel water level to the "low-low" level isolation setpoint. This resulted in a Group 1 isolation (automatic closure of all main steam line isolation valves) and initiation of the standby high pressure core cooling systems (HPCI and RCIC). After the vessel isolated one SRV lifted automatically as a result of the attendant continued pressure rise. During the period immediately following the scram and Group 1 isolation, vessel level and pressure were controlled by manual actuation of the safety-relief valves and manual control of the RCIC system. During this stabilization phase, it is believed that one of three SRVs selected for pressure control did not open as no indication, symptomatic of valve actuation, was apparent from the control room. At some time later, during this stabilization phase the drywell and reactor building air chillers tripped, for as yet unknown reasons. The absence of primary containment air cooling, coupled with the added containment heat loads associated with the discharging SRVs resulted in a rise in drywell pressure. The drywell reached the 2 psi Hi pressure setpoint at approximately 4:53 AM EST. This caused Group 2, 6 and 8 isolations which further isolated the drywell and inhibited reestablishment of normal drywell cooling. Over a period lasting approximately 4 to 5 hours the reactor was slowly brought to cold shutdown using the main condenser, while the containment was cooled down and depressurized via the standby gas treatment system and drywell vent system.

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During the period immediately following the reactor trip, several unanticipated systems interactions occurred in the reactor building outside primary containment. Following the initial scram on Hi-APRM, it is believed that the control room operator, in accordance with established procedures cleared and reset the scram signals so as to drain the scram discharge volume (SDV). During the period following the initial scram and vessel isolation, several additional reactor scrams and operator reset sequences apparently occurred as a result of reactor vessel level and pressure swings. The series of scrams, scram resets and SDV draining operations resulted in a considerable volume of relatively hot reactor water being discharged out of the SDV system. According to the resident at Hatch 2, water drained from the SDV headers is collected in a drain tank located in a corner room in the basement of the reactor building. Drains in the RCIC room are also connected to this sump. Apparently for some period following the initial scram a significant quantity of steam was able to back out of one or more of the connected RCIC room drains as a result of the rather high temperature scram discharge (reactor) water flowing into the drain tank. The temperature in the RCIC room rose sufficiently to set off one of the fire suppression sprinklers in the room. These sprinklers are designed to actuate at 165°F. Some of the sprinkler water fell onto hot RCIC steam supply piping components. This caused additional steam and vapor to be generated in the room. It is believed that the RCIC system did not trip or isolate as a result of the adverse environment.

As best as can be determined the RCIC room sprinkler system actuated a short time after the drywell reached 2 psig. After a considerable period of time a team was able to enter the RCIC room to secure the sprinkler system. During this time air in the area around the scram system hydraulic control units rose to about 130°F.

Cause and Corrective Action

Subsequent tests were performed on the main steam isolation valves. The tests showed steam passing through all of the steam lines except the "C" steam line with all valves indicating full open. The licensee concluded that the disk separated from the valve stem on either the inboard or outboard isolation valve initiating the transient. Both valves on the "C" steam line were shut, their operators disabled and tagged out of service. The reactor was subsequently brought back up to power and is currently being operated at about 80% power with only the A, B and D steam lines in operation.

Discussion

Previously, at Hatch 2, the "A" steam line inboard valve disk separated from the stem. This occurred on March 5, 1981. Numerous other similar occurrences have occurred at the Brunswick Units over the last several years. IE Notice No. 81-28 issued September 3, 1981 addressed these mechanical failures.

The fact that a considerable amount of hot reactor water flowed into the reactor building equipment drain tank as a result of the multiple reactor scrams and scram resets is not considered an unexpected phenomena. The fact that the adverse steam environment in the drain tank was able to be channeled back into the RCIC room through the connected RCIC room drain was unexpected. It is unclear whether reverse flow devices are not installed (either intentionally or unintentionally), or could not function for these conditions (i.e., for a steam/vapor medium). Should the licensee not take appropriate procedural or equipment-related corrective actions, it would appear that there would not be a good basis to conclude that a similar occurrence could not happen again.

Planned Further Actions

A teleconference call between AEOD (S. Rubin and E. Imbro) and Georgia Power has been requested and scheduled in order to collect additional information on the event, its cause, consequences and corrective actions. Further study of this event by AEOD (or its contractor) could address the design of the clean and dirty drain systems in this or other BWR reactor buildings. The purpose of such a study could be to assess the potential for this sort of common channeling back flow of hot liquid or steam vapor into the vital equipment areas. Such an investigation could be coupled with a survey of potential hot water sources inside the reactor building from both high and low energy piping systems. As a minimum, it is currently intended that the event will be written up for inclusion in Power Reactor Events. At the present time the licensee is expected to submit an LER on at least the initial main steam line isolation valve failure. It is requested that you support further investigation along these lines which would likely include a site visit to collect the needed information.

/s/

Stuart D. Rubin, Lead Engineer
Reactor Systems 4
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

cc: C. Michelson, AEOD
C. Heltemes, AEOD
M. El-Zeftawy, ROAB
J. Pellet, ROAB
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E. Imbro, ROAB
E. Brown, ROAB

Distribution:
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M. El-Zeftawy, ROAB
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C. Michelson, AEOD

PREDECISIONAL

OFFICE	ROAB	C/ROAB	~~AEOD~~	~~AEOD~~			
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DATE	9/20/82	9/24/82	~~8/ /82~~	~~9/ /82~~			