UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

October 6, 1995

Report Nos.: 50-321/95-15 and 50-366/95-15

Licensee: Georgia Power Company
P.O. Box 1295
Birmingham, AL 35201

Docket Nos.: 50-321 and 50-366 License Nos.: DPR-57 and NPF-5

Facility Name: Hatch Units 1 and 2

Inspection Conducted: August 21-25, 1995

Inspector: A. Gooden 10-06-95
for W. M. Sartor, Jr., Team Leader Date Signed

Accompanying Personnel: B. Haagensen, Consultant
A. Gooden, RII
J. Kreh, RII

Approved by: K. P. Barr 10/6/95
K. P. Barr, Chief Date Signed
Emergency Preparedness Section
Radiological Protection and Emergency Preparedness Branch
Division of Radiation Safety and Safeguards

SUMMARY

Scope:

This routine, announced inspection involved the observation and evaluation of the annual emergency preparedness exercise, conducted from 8:00 a.m. to 1:00 p.m. on August 23, 1995. The onsite inspection focused on the adequacy of the licensee's emergency response program, the implementation of the Emergency Plan and procedures in response to the simulated emergency conditions, and the effectiveness of the emergency response training program as reflected by the players' performance during the exercise. This exercise also included participation by NRC with the Executive Team in White Flint providing the Federal interface with the licensee during the earlier part of the exercise, and then turning the responsibility over to the NRC Region II Site Team that was co-located with the licensee. Correlative offsite activities involving State and local emergency response organizations were evaluated by the Federal Emergency Management Agency.

Results:

In the areas inspected, violations or deviations were not identified. The exercise demonstrated that the onsite emergency plans were adequate and that the licensee was capable of implementing them. An exercise strength was the licensee's self-evaluation/critique process which objectively identified, and presented to management, significant performance problems observed during the exercise. The performance problems addressed two fundamental and critical functional areas of emergency preparedness-accident classification and protective action decision making. The accident classification problem was the failure of the Emergency Director and his staff in the Technical Support Center to recognize the Site Area Emergency (SAE) (Paragraph 6). The protective action decision making issue was the failure of the Emergency Director in the Emergency Operations Facility to make protective action recommendations (PARs) consistent with the event classification (Paragraph 11). The licensee's overall performance during this exercise was considered to be satisfactory. The above issues were not identified as exercise weaknesses because the SAE declaration was made in a timely manner, and the PARs made were correct based on the PAR flowchart and the status of the containment barrier when the recommendations were made.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. Anderson, Unit Superintendent
*J. Bellmon, Nuclear Specialist
*C. Boone, Nuclear Specialist
*K. Breitenbach, Supervisor, Engineering
*C. Brown, Emergency Planning Coordinator (Corporate)
*C. Coop, Nuclear Specialist (Corporate)
*W. Eason, Supervisor, Safety Audit and Engineering Review
*P. Fornel, Manager, Maintenance
*D. Hart, Emergency Preparedness Coordinator
*W. Kirkley, Manager, Health Physics and Chemistry
*C. McDaniel, Acting Manager, Administration
*J. Payne, Senior Plant Engineer
*D. Read, Assistant General Manager, Plant Support
*R. Reddick, Emergency Preparedness Coordinator
*P. Roberts, Manager, Outage and Planning
*J. Robertson, Supervisor, Engineering Group
*D. Smith, Chemistry Superintendent
*L. Sumner, Nuclear Plant General Manager
*J. Thompson, Manager, Site Security
*S. Tipps, Manager, Nuclear Safety and Compliance

Other licensee employees contacted during this inspection included operators, engineers, security force members, and administrative personnel.

Other Personnel

*H. Dougherty, Oglethorpe Power Corporation

Nuclear Regulatory Commission

*J. Canady, Resident Inspector
*B. Holbrook, Senior Resident Inspector
*S. Vias, Region II

*Attended exit interview on August 25, 1995

An index of abbreviations used throughout this report will be found in the last paragraph.

2. Exercise Scenario (82302)

The scenario for the emergency exercise was reviewed to determine whether provisions had been made to test the integrated capability and a major portion of the basic elements existing within the licensee, State, and local emergency plans and organization as required by 10 CFR 50.47(b)(14), 10 CFR Part 50, Appendix E, Paragraph IV.F, and specific criteria in NUREG-0654, Section II.N.

The scenario was reviewed in advance of the scheduled exercise date and was discussed with licensee representatives. The scenario developed for this exercise was adequate to fully exercise the onsite and offsite emergency organizations of the licensee and provided sufficient emergency information to the State and local government agencies for their full participation in the exercise. While no major problems with the scenario were identified during the review, an inconsistency regarding the status of the unaffected unit was observed to cause some confusion during the exercise. This inconsistency occurred as a result of a controller providing information on the unaffected unit to a player that was different from the initial conditions provided in the scenario. This issue was considered to be an exercise control problem and did not detract from the overall performance of the licensee's emergency organization.

A narrative description of the scenario follows:

The exercise scenario started at 8:00 a.m. with a 64 gpm LOCA into the drywell from a CRD housing failure. The SOS declared an alert and activated the emergency plan.

After the TSC and OSC were activated, a lightning strike caused the loss of the 2C SUT. Shortly after the loss of the 2C SUT, the 2D SUT failed and a loss of 2 of 3 unit emergency buses occurred after the associated EDGs failed to start. The remaining EDG (2C) was receiving cooling water from the 2B PSW which tripped due to a control relay failure. The operators stopped the remaining EDG prior to exceeding design temperatures. This caused a Unit 2 blackout and escalation to the SAE classification.

An emergency depressurization of the reactor vessel was directed when drywell temperatures approached limiting values due to the LOCA. This caused a rapid reduction in RPV level and soon reduced RPV pressure below the minimum required pressure for RCIC operation.

The 2C EDG was repaired and restarted to power the 2G 4160 V emergency bus. The associated LPCI and core spray pumps failed to start. The reactor had no operable sources of injection because the HPIC system was out of service for maintenance and RCS pressure was too low for RCIC operation. With a LOCA in progress and no sources of injection, core uncovery subsequently occurred causing clad damage and subsequent fuel melt. A release began when design leakage from the drywell caused elevated levels in the Reactor Building and the SBGT system exhausted this material to the environment through the stack. Classification was escalated to a GE based upon failure of 2 of 3 fission product barriers (clad and RCS) with the potential for the failure of containment. Protective actions were recommended to evacuate the 2 mile radius and 5 mile downwind sectors while sheltering the remainder of the 10 mile EPZ.

The 2A EDG was repaired and restarted. The 2E emergency bus was repowered and the 2A LPCI pump was started reflooding the core. After core recovery, the drywell purge line ruptured outside of containment causing an 18 inch release path into the Reactor Building. The release rate significantly increased as SBGT continued to exhaust the Reactor Building elevated source term into the stack. This event significantly increased the release rate causing peak offsite dose rates up to 94 mr/hr. The PARs were extended to evacuate the 5 mile radius and 10 mile downwind sectors while sheltering the remaining EPZ.

The exercise terminated at 1:00 p.m. with the reactor in the process of being placed into cold shutdown.

No violations or deviations were identified.

3. Assignment of Responsibility (82301)

This area was observed to determine whether primary responsibilities for emergency response by the licensee had been specifically established and whether adequate staff was available to respond to an emergency as required by 10 CFR 50.47(b)(1) and 10 CFR Part 50, Appendix E, Paragraph IV.A.

The inspectors observed that specific assignments had been made for the licensee's ERO and that there was adequate staff available to respond to the simulated emergency. The initial response organization was augmented by designated licensee representatives. The capability for long-term or continuous staffing of the ERO was discussed, and planning for relief was initiated at each of the ERFs.

No violations or deviations were identified.

4. Onsite Emergency Organization (82301)

The licensee's onsite emergency organization was observed to determine whether the responsibilities for emergency response were defined, whether adequate staffing was provided to insure initial facility accident response in key functional areas at all times, and whether the interfaces were specified as required by 10 CFR 50.47(b)(2) and 10 CFR Part 50, Appendix E, Paragraph IV.A.

The inspectors determined that the licensee's onsite emergency organization was well defined and was generally effective in dealing with the simulated emergency. Adequate staffing of the ERFs was provided for the initial accident response, and the interfaces between the onsite organization and offsite support agencies were adequate to ensure prompt notification and support from offsite agencies as required.

No violations or deviations were identified.

5. Emergency Response Support and Resources (82301)

This area was observed to determine whether arrangements for requesting and effectively using assistance resources were made, whether arrangements to accommodate State and local personnel in the EOF were adequate, and whether other organizations capable of augmenting the planned response were identified as required by 10 CFR 50.47(b)(2) and 10 CFR Part 50, Appendix E, Paragraph IV.A.

The licensee's Emergency Plan provided information regarding additional support and resources that may be called upon to assist in an emergency. The inspector observed that representatives of the State of Georgia was readily accommodated at the EOF.

No violations or deviations were identified.

6. Emergency Classification System (82301)

This area was observed to verify that a standard emergency classification and action level scheme was in use by the licensee as required by 10 CFR 50.47(b)(4) and Paragraph IV.C of Appendix E to 10 CFR Part 50.

An inspector observed that the emergency classification system was in effect as stated in the Emergency Plan and procedures. Although the system appeared to be adequate for the classification of the simulated accident, the ED and his TSC staff failed to recognize the initiating condition for the SAE classification. Specifically, the TSC team did not recognize that a loss of all AC power on a single unit should have been classified at the SAE level. The team misread the EAL and determined that both units had to lose power before the declaration of a SAE was appropriate. The ED directed that the station remain at the Alert classification. Shortly after this decision, the EOF correctly recommended escalation to a SAE and after some discussion, this recommendation was properly implemented by the ED in the TSC. The licensee identified the failure of the ED and his TSC staff to recognize the SAE classification as an item requiring corrective action in their critique process.

There was also confusion and uncertainty regarding the GE classification that was made by the ED in the EOF. Specifically, after talking with the NRC Executive Team, the ED decided to upgrade to a GE even though he felt it was conservative and was not required by the Hatch EALs. The reason for the different assessment of the emergency classification between the NRC and the Plant Hatch ED was due to a difference between Section 22.5 of the Hatch EALs and NUREG-0654 example initiating conditions as follows:

"22.5 FAILURE OF CLADDING AND PRIMARY COOLANT BOUNDARY WITH POTENTIAL LOSS OF CONTAINMENT
Emergency conditions exist when:
A failure of cladding and primary coolant boundary <u>WITH</u> potential loss of containment is indicated by any <u>one</u> condition being met <u>from each of the following three subsections (22.5.1, 22.5.2, and 22.5.3):"</u> ...

(Note: *Sections 22.5.1 and 22.5.2 were clearly met by barrier failure of clad and RCS*)

"22.5.3 <u>Containment</u>
A <u>potential loss of containment</u> exists <u>IF</u> any of the following exists:

22.5.3.1 Loss of Primary Containment Cooling capability <u>WITH</u> primary containment temperature <u>AND/OR</u> pressure increasing <u>AND</u> approaching 56 psig

-<u>OR</u>-

22.5.3.2 Drywell or torus hydrogen concentration $\geq$ 6% <u>with</u> Drywell or Torus Oxygen Concentration $\geq$ 5%

-<u>OR</u>-

22.5.3.3 SOS/ED judgement that the loss of containment is imminent**

(**) IF a parameter is approaching emergency action level criteria and mitigation systems are unavailable, assume the barrier will be lost."

This EAL focused the ED's decision on two conditions: the probability of exceeding the design pressure or explosive mixture concentrations in the drywell. The EAL does not refer to other possible precursors for containment failure mechanisms such as a LOCA with a failure of ECCS which are explicitly stated in the NUREG-0654 example initiating conditions.

The ED did not conclude that plant conditions warranted imminent failure of containment because the Hatch EAL limited his consideration of alternative precursor indications of containment failure. The NRC Executive Team used the NUREG-0654 example initiating conditions for the basis for their classification recommendations. The classification as made by the ED was conservative in accordance with the Hatch EALs.

No violations or deviations were identified.

7. Notification Methods and Procedures (82301)

This area was observed to determine whether procedures had been established for notification by the licensee of State and local response organizations and emergency personnel; whether the content of initial and follow-up messages to response organizations had been established; and whether means to provide early notification to the populace within the plume exposure pathway EPZ had been established as required by 10 CFR 50.47(b)(5) and 10 CFR Part 50, Appendix E, Paragraph IV.D.

An inspector observed that notification methods and procedures had been established and were used to provide information concerning the simulated emergency conditions to Federal, State and local response organizations and to alert the licensee's augmented emergency response organization. Generally, the notifications were timely and provided an adequate description of the emergency conditions. One exception to the latter was the description which accompanied the GE declaration. The description provided did not support a GE declaration; however, the required PARs were made. A second exception was the time required to provide revised PARs (Paragraph 11).

No violations or deviations were identified.

8. Emergency Communications (82301)

This area was observed to determine whether provisions existed for prompt communications among principal response organizations and emergency personnel as required by 10 CFR 50.47(b)(6); 10 CFR Part 50, Appendix E, Paragraph IV.E; and specific criteria in NUREG-0654, Section II.F.

Communications among the licensee's emergency response facilities and emergency organization and between the licensee's emergency response organization and offsite authorities were good. No communications related problems of significance were identified during the exercise.

No violations or deviations were identified.

9. Public Education and Information (82301)

This area was observed to determine whether information concerning the simulated emergency was made available for dissemination to the public as required by 10 CFR Part 50, Appendix E, Paragraph IV.D, and specific criteria in NUREG-0654, Section II.G.

The licensee established its Emergency News Center to demonstrate timely and accurate response to news inquiries. This area was not evaluated by the NRC inspection team.

No violations or deviations were identified.

10. Emergency Facilities and Equipment (82301)

This area was observed to determine whether adequate emergency facilities and equipment to support an emergency response were provided and maintained as required by 10 CFR 50.47(b)(8); 10 CFR Part 50, Appendix E, Paragraph IV.E; and specific criteria in NUREG-0654, Section II.H.

The inspectors observed the activation and staffing of the Simulator Control Room, TSC, OSC, and the EOF. Selected aspects of the operation of these facilities was observed and no facility or equipment issues were identified that hampered the emergency response.

No violations or deviations were identified.

11. Protective Responses (82301)

This area was observed to determine whether guidelines for protective actions during the emergency, consistent with Federal guidance, were developed and in place, and protective actions for emergency workers, including evacuation of nonessential personnel, were implemented promptly as required by 10 CFR 50.47(b)(10) and specific criteria in NUREG-0654, Section II.J.

An inspector noted that the licensee had implementing procedures for formulating PARs for the offsite populace within the 10-mile EPZ. The licensee's PARs were consistent with the EPA criteria; however, they were not consistent with the basis for the GE declaration. As noted in Paragraph 6, the ED, after talking with the NRC, declared the GE on the basis of multiple barrier failures; specifically the RCS and clad had failed and that containment failure was imminent. However, when conferring with his Dose Assessment Manager, the ED concurred in the PARs based on the decision that containment failure was not imminent. The PAR made was "Recommend evacuation 0-2 mile radius, 2-5 miles downwind. Shelter remainder 2-5 miles, 5-10 miles downwind." The protective action recommendation that was consistent with the basis for the GE declaration was "Recommend evacuation 0-5 miles complete radius, 5-10 miles downwind. Shelter remainder of the EPZ." The licensee identified the failure to make the PAR consistent with the basis for the GE declaration as an item requiring corrective action. The inspector also noted that it took the licensee 30 minutes to release the message to the State and counties following the decision to upgrade the PARs to the more conservative ones that were consistent with the emergency classification.

Procedures for protective actions for emergency workers, including the evacuation of nonessential personnel, were available but were not demonstrated as an objective for this exercise.

No violations or deviations were identified.

12. Exercise Critique (82301)

The licensee's critique of the emergency exercise was observed to determine whether deficiencies identified as a result of the exercise and weaknesses noted in the licensee's emergency response organization were formally presented to licensee management for corrective actions as required by 10 CFR 50.47(b)(14); 10 CFR Part 50, Appendix E, Paragraph IV.E; and specific criteria in NUREG-0654, Section II.N.

The licensee conducted player critiques following termination of the exercise. A detailed controller/evaluator critique was conducted on the day after the exercise. A formal presentation of the licensee's critique conclusions was made on August 25, 1995. The licensee's critique process was considered as a strength as they proactively identified for corrective actions the problems addressing accident classification and PAR decision making.

No violations or deviations were identified.

13. Exit Interview

The inspection scope and results were summarized on August 25, 1995, with the persons whose names are listed in Paragraph 1. The Team Leader described the areas inspected and discussed observations made during the inspection. Licensee management was informed that the NRC considered the exercise to have been successful in spite of the problems of accident classification and PAR decision making because the licensee had identified these areas for corrective action. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

14. Federal Evaluation Team Report

The report by the Federal Evaluation Team (Regional Assistance Committee and Federal Emergency Management Agency, Region IV) concerning the activities of offsite agencies during the exercise will be forwarded by separate correspondence.

15. Index of Abbreviations Used in This Report

AC	Alternating Current
CFR	Code of Federal Regulations
CRD	Control Rod Drive
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ED	Emergency Director
EDG	Emergency Diesel Generator
EOF	Emergency Operations Facility
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
ERF	Emergency Response Facility
GE	General Emergency

GPM	Gallons Per Minute
ERO	Emergency Response Organization
HPIC	High Pressure Injection Coolant
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
mr/hr	Millirem Per Hour
NRC	Nuclear Regulatory Commission
OSC	Operational Support Facility
PAR	Protective Action Recommendation
PSW	Plant Service Water
RCIC	Reactor Core Isolation Coolant
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SAE	Site Area Emergency
SBGT	Standby Gas Treatment
SOS	Shift Operations Supervisor
SUT	Start Up Transformer
TSC	Technical Support Center

Attachment (10 pages):
Objectives, Narrative Summary, and Timeline
for Plant Hatch 1995 Exercise

2.2 OBJECTIVES

The E. I. Hatch Nuclear Plant 1995 emergency preparedness exercise objectives are based upon Nuclear Regulatory Commission requirements provided in 10 CFR 50, Appendix E, ***Emergency Planning and Preparedness for Production and Utilization Facilities***. Additional guidance provided in NUREG-0654, FEMA-REP-1, Revision 1, ***Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants***, was utilized in developing the objectives. The following objectives for the exercise are consistent with the aforementioned documents:

A. Accident Assessment and Classification

1. Demonstrate the ability to identify initiating conditions, determine Emergency Action Level (EAL) parameters and correctly classify the emergency throughout the exercise.

2. Demonstrate the ability to provide core damage assessments.

B. Notification

1. Demonstrate the ability to alert, notify and mobilize appropriate station and corporate emergency response personnel.

2. Demonstrate the ability for prompt notification of the State, Local and Federal authorities.

3. Demonstrate the ability to warn or advise onsite individuals (including employees, visitors and contract personnel) of an emergency condition.

C. Emergency Response

1. Demonstrate that an individual is assigned and is in charge of the emergency response.

2. Demonstrate planning for 24-hour per day emergency response capabilities.

3. Demonstrate the line of succession for the Emergency Director.

4. Demonstrate timely response of station and corporate management, administrative and technical staff.

5. Demonstrate the timely activation of the Technical Support Center (TSC), Operations Support Center (OSC), and Emergency Operations Facility (EOF).

6. Demonstrate the adequacy of equipment, security provisions and habitability precautions for the TSC, OSC, EOF, GOOC and the Emergency News Center (ENC).

7. Demonstrate satisfactory communications ability of all emergency support resources.

2.2 OBJECTIVES

D. Radiological Assessment and Control

1. Demonstrate the coordinated gathering of radiological and non-radiological (meteorological) data necessary for emergency and environmental response including collection and analysis of in-plant surveys and samples.

2. Demonstrate the ability to develop dose projections, compare the projections to Protective Action Guidelines (PAGs) and determine and recommend the appropriate protective actions.

3. Demonstrate onsite contamination control measures including area access control.

4. Demonstrate the ability for determining projected doses from available plant instrumentation.

5. Demonstrate onsite PAGs for a select number of personnel, as appropriate.

6. Demonstrate the decision making process for authorizing emergency workers to receive radiation doses in excess of Plant Hatch administrative limits, as appropriate.

7. Demonstrate the ability for collection and analysis of Post-Accident samples.

E. Public Information Program

1. Demonstrate the timely and accurate response to news inquiries.

2. Demonstrate timely preparation of accurate news releases and suppression of rumors.

3. Demonstrate the adequacy of the Emergency News Center (ENC).

F. Evaluation

1. Demonstrate ability to conduct a post-exercise critique to determine areas requiring additional improvement.

Areas of the HNP Emergency Plan that will NOT be demonstrated during this exercise include:

- Onsite personnel evacuation
- Actual shift turnover (long term shift assignments will be demonstrated by rosters).
- Real time actuation of the ENC
- Conduct of reentry and recovery operations

EDWIN I. HATCH NUCLEAR PLANT
1995 EMERGENCY PREPAREDNESS
EVALUATED EXERCISE

3.1 NARRATIVE SUMMARY

This scenario begins with a routine control rod exercise surveillance in accordance with (IAW) procedure 34SV-C11-003-2S, ***Control Rod Weekly Exercise***. When rod #26-03 is selected for exercise, it will not insert and will uncouple if any withdrawal motion is selected. The exercise of this rod will coincide with an unidentified primary system leakage into the drywell from the #26-03 Control Rod Drive (CRD) mechanism housing to reactor vessel flange as a result of cracked and failing CRD housing cap screws. As a result of the unidentified reactor coolant system (RCS) leakage >50 gpm, an **ALERT** should be declared IAW procedure 73EP-EIP-001-0S, ***Emergency Classification and Initial Actions***, Section 20.0, "Loss of Coolant".

Operators should begin venting the drywell through the Standby Gas Treatment System (SBGTS) IAW procedure 34AB-T23-002-2S, ***Small Pipe Break Inside Primary Containment***, to control containment pressure. Operators should enter emergency operations procedures on Primary Containment Control criteria as drywell pressure and temperature increase. A controlled reactor shutdown should commence IAW Procedure 34GO-OPS-014-2S, ***Fast Reactor Shutdown***. The leakrate will remain relatively constant throughout the shutdown process as a function of RCS pressure.

A rapidly moving cold front passes through the area bringing heavy rains high wind gusts and thunderstorms. Lightning strikes the 2D start-up transformer (SUT) resulting in a fault. Fire suppression systems function as designed to extinguish the resulting fire. Dispatch of the fire brigade is not required. Loss of the 2D SUT will deenergize all 4KV emergency busses. They will be reenergized from the remaining 2C SUT. The 2C Emergency Diesel Generators (EDG) will autostart and run unloaded. Pre-existing deposits on the fuel control racks on the 2A EDG will cause binding of the fuel control racks and prevent the start-up of the 2A EDG. Damage assessment teams should be dispatched to determine the extent of damage to the 2D SUT and to investigate problems with the 2A EDG.

Moments later, the support conductor for the 2C SUT, damaged during the initial lightning strike, breaks free from the turbine building and drops to the ground.. This results in the loss of the 2C SUT and its associated electrical loads. Available emergency diesel generator 2C will energize the 2G 4KV emergency buses.

3.1 NARRATIVE SUMMARY

Due to a faulty relay, the 2B PSW pump experiences a trip. Failed system logic on the 2C EDG will trip the diesel when jacket cooling temperature reaches 205 degrees. Alternatively, the operators may secure the diesel, recognizing the significant potential for damage to the only available EDG (i.e., loss of cooling water). Either case will result in a total loss of offsite power for the plant and subsequent declaration of a **SITE AREA EMERGENCY (SAE).** Repair teams should be dispatched to the intake structure to repair the faulty relay on the 2B PSW pump.

Shortly after the replacement of the faulty relay in the PSW pump, the leaking CRD housing cap screw fails and the CRD housing separates from the vessel flange. The #26-03 rod is mechanically bound in its channel and does not seat onto the thermal sleeve, resulting in an RCS LOCA of approximately 1000 gpm. Reactor water level begins to drop dramatically.

Low Pressure Coolant Injection (LPCI) pump 2B autostarts with power supplied from the 2G 4KV emergency bus. LPCI injection valve 2E11-F015B fails to open due to a hydraulic locking condition in its Limitorque motor operator. This prevents injection with this train of LPCI, but will still allow torus cooling operation. An alternate injection path may be accomplished through the opening of the manually isolated deenergized cross-connect valve F010 and de-energized "A" train injection valve 2E11-F015A. Core Spray (CS) pump 2B fails to autostart with power supplied from the 2G 4KV emergency bus due to a problem with the pump motor windings.

With no High Pressure Coolant Injection (HPCI) feed to the reactor vessel (i.e., HPCI turbine is tagged-out for repair) and the ongoing CRD housing LOCA, the reactor vessel depressurizes and drains down to the drywell.

3.1 NARRATIVE SUMMARY

As the fuel becomes uncovered, fuel cladding failure begins and gas gap activity is released to the coolant and containment. Within approximately thirty minutes, fuel decay heat has produced fuel damage releasing non-gap fission products to the coolant and containment. As the released activity is transported outside the sacrificial shield in RCS piping and released to the containment atmosphere through the CRD LOCA, drywell post accident radiation monitors trend upward. As the released activity is transported to the torus by water drainage down the risers, torus post-LOCA radiation monitors trend upward and radiation doses from the torus and active ECCS system piping begin to impact repair team activities in the lower Reactor Building areas.

Successful repair of the 2A EDG fuel control racks will allow the recovery of the 2A EDG and re-energization of the 2E 4KV emergency bus. This will allow the starting of the 2A Core Spray pump and the 2A RHR pump.

Fuel damage is halted at approximately 90% gas gap release and 15% fuel melt. Shortly after the recovery of RCS feed, the thermal and hydraulic shock of the restored feed causes rod #26-03 to release from its stuck position and seat on the thermal sleeve, stopping the LOCA to the containment. Core Spray continues to cool the core and LPCI 2A provides feed, flooding-up and recovering the exposed fuel.

The 18" drywell purge line suffers a complete shear upstream of 2T48-F319. A **GENERAL EMERGENCY** should be declared IAW procedure 73EP-EIP-001-0S, ***Emergency Classification and Initial Actions***, Section 22.0, "Multiple symptoms and other conditions".

3.1 NARRATIVE SUMMARY

Equalization of drywell pressure to atmospheric pressure will terminate the release. Release of radioactivity to the environment will decline as the SBGTS "turns-over" the Reactor Building atmosphere.

After field monitoring teams have tracked and monitored the radiological release and protective action considerations have been determined and recommended from the EOF and Emergency Operations Centers (EOCs), the exercise is terminated.

3.2 SCENARIO TIMELINE

TIME (EASTERN)	EVENT DESCRIPTION
0730	Initial Conditions:

HNP Unit 2 is in End of Core Life (EOL) at 100 % power. Reactor power has been ≥ 90% power for the last 180 days and Xenon is at equilibrium. Unit 1 is in normal full-power operation. The 2A EDG has been tested and returned to service following repair of the Woodward governor

The following equipment is out-of-service:

- The backup meteorological tower suffered a lightning strike the previous day and is no longer providing data feed. It is under inspection and repair.
- The "Swing" Emergency Diesel Generator (EDG) 1B is under repair for a damaged crankshaft bearing identified on the last routine surveillance operation. The EDG is being disassembled and work is anticipated to require another 24 hours to complete. This places both Units in a 72 hour Limiting Condition for Operation (LCO) and the clock began at 2200 yesterday when the 1B EDG was tagged-out.
- The High Pressure Coolant Injection (HPCI) steam turbine is tagged-out for the repair of the turbine-driven oil pump. Low and irregular oil pressure resulting in governor tripping was occurring in test operation. Pump overhaul parts are on-hand. The HPCI turbine was tagged-out at 0100 today and disassembly began on the night shift. This places Unit 2 in another LCO of 14 days.

Weekly control rod exercising in accordance with procedure 34SV-C11-003-2S, ***Weekly Control Rod Exercise***, is to be performed this morning.

Plant radiological and radiochemical conditions are normal.

The winds are from the east at 3 to 4 mph. Temperature is 76° with an anticipated high in the low nineties. A severe thunderstorm warning has been issued by the National Weather Service for Toombs, Appling, Jeff Davis and Tattnall Counties. This warning is in effect until 1030 today.

3.2 SCENARIO TIMELINE

TIME (EASTERN) EVENT DESCRIPTION

0730 (continued)

After assuming control of the Unit 2 control room (Simulator) and walking-down the panels, operators should maintain steady-state operation and begin the weekly control rod exercise in accordance with (IAW) procedure 34SV-C11-003-2S, Weekly Control Rod Exercise.

0745

When rod #26-03 is selected for testing, it will not move in the "insert" direction. If "withdraw" is selected on the rod control switch at any time during this test, control rod drive (CRD) overtravel alarm and indication will occur indicative of an uncoupled rod.

0800

A leakage of the reactor coolant system (RCS) to the drywell (DW) will begin from the #26-03 CRD flange at approximately 60 gpm.

0815

DW temperature, humidity and pressure will begin to rise. DW floor drain and equipment drain sumps will alarm. Sump timers will start. Radwaste (RW) sump flow integrators data will indicate 60 gpm leakage to the DW. An **ALERT** should be declared based upon >50 gpm unidentified RCS leakage (73EP-EIP-001-OS, Emergency Classification and Initial Actions, section 20.0, "Loss of Coolant"). Drywell venting should begin through the Standby Gas Treatment System (SBGTS) to control DW pressure. A controlled quick shutdown of the reactor should commence.

0845

A thunderstorm passes through the area. Lightening strikes the 2D start up transformer (SUT). Loss of the 2D SUT and its associated electrical loads occurs. Fire suppression systems actuate to control the event. Power lineup auto transfers to the 2C SUT alternate supply and the electrical system is stabilized. The 2C Emergency Diesel Generator (EDGs) autostarts and runs unloaded. The 2A EDG fails to start due to binding of the fuel control racks.

0900

The conductor support for the 2C SUT, damaged during the initial lightning strike, breaks free from the turbine building and drops to the ground. A ground fault and subsequent loss of the 2C SUT occurs. The 2C EDG loads on the loss of 4KV bus voltage.

3.2 SCENARIO TIMELINE

TIME (EASTERN) EVENT DESCRIPTION

0900 (continued)

The 2B plant service water (PSW) pump trips due to a faulty relay. The remaining 2C EDG begins to overheat due to a loss of cooling capacity. Failed system logic will trip the diesel when jacket cooling temperatures reach 205 degrees. This will occur prior to mechanical seizure of the diesel Plant personnel may secure the 2C EDG to prevent the diesel from seizing. Either action will result in a loss of all onsite power.

0930

Loss of all onsite power coupled with the loss of all offsite power will result in the declaration on a **SITE AREA EMERGENCY (SAE)**.

1000

The faulty relay in the 2B PSW pump has been replaced. The 2C EDG may be restarted at this time.

1020

The leaking #26-03 CRD housing separates from the reactor lower head flange. The RCS leak is now a loss of coolant accident (LOCA) of approximately 1,000 GPM. As the reactor water level and reactor pressure decrease, multiple failures in low pressure systems prevent adequate feed to maintain reactor water level.

RHR pump 2B autostarts off the 2G 4KV bus but inboard injection valve 2E11-F015B fails in the closed position, preventing feed to RCS without manual opening of the RHR cross-connect valve (2E11-F010) and "A" train injection valve 2E11-F015A. The RHR to PSW crosstie valve (2E11-F073B) fails to open due to a bent stem.

Core Spray (CS) pump 2B fails to autostart off the 2G 4KV bus due to problem with the motor windings.

RCIC autostarts but experiences spurious trips.

With no available emergency core cooling system (ECCS) feed to the RCS, the reactor vessel depressurizes and begins to drain down. Drywell temperature and pressure increase as the RCS drains into the drywell.

3.2 SCENARIO TIMELINE

TIME (EASTERN)	EVENT DESCRIPTION

1040

As the uncovered fuel overheats from decay heat, gas gap activity and fuel damage releases radioactive material to the vessel. Released activity is drained to the Drywell and Torus through the CRD LOCA. Drywell and Torus radiation monitors increase. The source term builds up in the torus and running torus cooling piping of the active RHR system, reducing access of repair teams to the 2E11-F015B valve and the RHR cross-connect valves.

1100

EDG 2A fuel control racks are freed. EDG 2A may be started and loaded onto the 2E 4KV bus.

Core Spray 2A and RHR 2A auto start and begin feeding the reactor vessel.

1105

The mechanical and thermal shock of the restored feed to the reactor vessel causes stuck control rod #26-03 to release from its stuck position and seat onto the thermal sleeve, effectively stopping the CRD LOCA flow to the drywell.

1115

The 18 inch line from the drywell to SBGT shears at the weld union upstream of 2T48-F319. Area Radiation Monitors (ARMs) monitors alarm as the radioactivity is liberated to the reactor building. As the drywell depressurizes into the reactor building, radioactive material is diffused into the reactor building atmosphere and pumped to the environment through the SBGTS.

A **GENERAL EMERGENCY** should be declared based on loss of containment (*73EP-EIP-001-OS, Emergency Classification and Initial Actions, section 22.0, "Multiple symptoms and other conditions"*).

1230

The primary system has been flooded-up, cooled-down and stabilized. Release to the environment is declining as SBGTS turns over the contaminated atmosphere of the reactor building. The release has been monitored, tracked and sampled.

The exercise is terminated.