ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-285/95-15

License: DPR-40

Licensee: Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Hwy. 75 - North of Fort C
Fort Calhoun, Nebraska

Facility Name: Fort Calhoun Station

Inspection At: Fort Calhoun, Nebraska

Inspection Conducted: November 13-16, 1995

Inspectors: Arthur D. McQueen, Emergency Preparedness Analyst
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Approved: Blaine Murray 11/29/95
Blaine Murray, Chief, Plant Support Branch
Division of Reactor Safety

Date

Inspection Summary

Areas Inspected: Routine, announced inspection of the licensee's performance and capabilities during the full-scale exercise of the emergency plan and implementing procedures. The inspection team observed activities in the control room (simulator), technical support center, operations support center, and emergency operations facility.

Results:

- The control room staff's performance during the exercise was good. Excellent communication and repeat-backs were used. The shift supervisor was knowledgeable of the plant conditions and held frequent meetings to brief the control room staff on these conditions. The control room evaluated plant conditions and made the proper emergency classification consistent with plant conditions (Section 2).

- At the technical support center, technical assessment of plant conditions and contingency measures was generally excellent. Engineering personnel and the reactor safety coordinator were continually looking ahead and formulating contingencies and recommendations based on known plant conditions (Section 3).

- The failure to have an individual properly trained and capable of performing core uncovery predications necessary to evaluate plant conditions was identified as an exercise weakness (Section 3).

- The operations support center was activated promptly and was proactive in pursuing repair priorities. Communication between the operations support center, the inplant emergency teams, and other emergency response facilities were frequent and effective (Section 4).

- The operations support center was efficiently readied for communication and for team formation and dispatch. Communication between the operations support center director and support staff, technical support center, and control room simulator personnel was very good and effective in carrying out the duties of the operations support center (Section 4).

- The degree of simulation of several activities by in-plant response teams during the exercise precluded the evaluation of individuals' knowledge of duties and was identified as an exercise weaknesses (Section 4).

- Command and control in the emergency operations facility were generally good. Notifications to offsite response agencies of the general emergency and appropriate protective action recommendations were made in a timely manner and, generally, included appropriate information (Section 5).

- Upon their arrivals, offsite representatives from the states of Iowa and Nebraska were quickly and effectively integrated into the emergency response organization. Interactions with offsite officials, a mock NRC-site team, and other organizations were excellent (Section 5).

- The licensee conducted staggered shift changes of staff at the technical support center, the operations support center and the emergency operations facility. They were considered effective and an exercise strength (Section 6).

- The exercise scenario provided sufficient challenges to test emergency response capabilities and demonstrate exercise objectives (Section 6).

- The licensee's critique process was effective (Section 7).

Summary of Inspection Findings:

- Exercise Weakness 285/95015-01 was opened (Section 3).
- Exercise Weakness 285/95015-02 was opened (Section 4).

Attachments:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Licensee Scenario Summary and Timeline

DETAILS

1 PROGRAM AREAS INSPECTED (82301)

The licensee's emergency exercise began at 7 a.m. on November 14, 1995. The licensee activated its emergency response organization and all emergency response facilities. Offsite participation in this biennial, full-participation exercise included the states of Iowa and Nebraska.

The scenario for the exercise was dynamically simulated using the Fort Calhoun Nuclear Station control room simulator. A licensee summary of the exercise scenario, including major events and a timeline is attached (Attachment 2).

2 CONTROL ROOM (82301-03.02)

The inspectors observed and evaluated the control room staff as they performed tasks in response to exercise events indicated by the control room simulator. These tasks included detection and classification of event-related conditions, detailed analysis of conditions, notification of licensee personnel, and notification of offsite authorities.

At 7:34 a.m., Fire Detectors FD-64-03, 04, 05, and 06 all alarmed and Diesel Generator 2 deluge system actuated. This caused a fire main low pressure alarm and an automatic start of the motor-driven fire pump. The deluge system extinguished the fire at 7:58 a.m. Shortly thereafter, and on repeated occasions, the control room requested the hazardous material team to obtain air samples of Room 64 to allow personnel to enter the room and assess the damage to the diesel generator. The samples were finally obtained and reported to the control room at 8:29 a.m. The samples did not indicate any toxic conditions.

At 7:35 a.m., the control room entered Abnormal Operating Procedure AOP-06, "Fire Emergency," and made an emergency announcement over the plant paging system requesting that fire brigade personnel report to the designated assembly area and all other personnel stay clear of the affected area. This announcement was inaudible in certain areas of the plant and had to be repeated. At 7:39 a.m., control room personnel notified the Blair fire department and requested outside assistance. At 7:44 a.m., fire brigade personnel were dispatched to fight the fire and at 7:45 a.m. the licensee declared an "Alert" based on Emergency Action Level EAL 6.2, "Fire Affecting One Train of ESF." The licensee made all the required notifications to federal and state agencies within the required time periods for this, and every other, emergency classification. At 8:03 a.m. the emergency response organization was notified.

The shift supervisor decided at 7:52 a.m. to evacuate all nonessential personnel from the site. The evacuation alarm was sounded at 7:54 a.m. Based on the guidance contained in Attachment 6.2 of Emergency Plan Implementing Procedure EPIP-OSC-2, "Command And Control Position Actions/Notifications,"

the shift supervisor directed plant personnel to evacuate the site. The north security access point and proceed to the administration building. The decision on which egress point to use was primarily based on wind direction. However, in following this guidance, plant personnel would have egressed through a toxic gas plume if toxic gases had been generated during the diesel generator fire because the exhausts from the diesel generator were being blown directly into the egress path. The control room did not receive positive indication that toxic gases were not being produced until after the evacuation began. The site evacuation began at 7:54 a.m. and the control room was not made aware of the results of air samples taken by the hazardous material team until 8:29 a.m. Also, in using the guidance provided in Attachment 6.2, if a radiation release had occurred during the scenario, plant personnel would have evacuated through a radiation plume. The inspectors later learned that the licensee had identified on October 30, 1995, that the guidance provided in Attachment 6.2 was incorrect and a procedural change had been initiated to correct these errors.

At 7:57 a.m., the shift supervisor made a second evacuation announcement. During this announcement the shift supervisor announced the wrong assembly area. This was caused by an error in the licensee's procedure which listed the wrong assembly location. After realizing the mistake, the correct announcement was immediately made. The inspectors were later informed that this was due to an outdated instruction in the simulator.

At 8:40 a.m., command and control was transferred to the technical support center. The announcement that transferred command and control was not heard by all control room personnel.

An earthquake occurred at 9 a.m. that lasted for approximately 40 seconds. The licensed senior operator made a conservative decision to manually trip the reactor after the earthquake. Diesel Generator 1 started but tripped on overspeed. Investigation determined an oil line was broken on the inlet to the governor booster pumps. The diesel started with the fuel racks in the full fuel position with the governor inoperative due to loss of oil. The earthquake caused Valve HCV-348, "Shutdown Cooling Suction Valve," located inside containment to go to the intermediate position and cracked the pipe downstream of Valve HCV-347, Shutdown Cooling Suction Valve," located outside containment. The shutdown cooling relief valve cycled open and relieved to the pressurizer quench tank (76 gallons per minute), eventually causing the rupture disk to rupture. A 24 gallons per minute leak also occurred outside containment as through the cracked pipe associated with Valve HCV-347. This resulted in a loss-of-coolant accident inside the auxiliary building. Additionally, the roll-up door between the auxiliary building and the radioactive waste building was damaged and partially blocked open. Numerous windows in the radioactive waste building and maintenance shop were also broken. The control room was not made aware of the damage to the roll-up door or the broken windows. This constituted a potential release path.

A "Site Area Emergency" was declared at 9:07 a.m., based on Emergency Action Level EAL 11.12, "Plant Condition Warrant Activation of All Emergency Facilities."

A second earthquake occurred at 10:05 a.m. that lasted for approximately 25 seconds. The earthquake caused a loss of 345 kV and 161 kV offsite power. All power to safeguards busses was lost.

At 10:10 a.m., the reactor coolant system leak rate through Valve HCV-348 increased to approximately 1000 gallons per minute. A "General Emergency" was declared based on Emergency Action Level EAL 1.19, "Imminent Core Uncovery With Containment Failure or Challenge." Also, the control room coordinator invoked the 10 CFR 50.54(x) provision to exceed the Technical Specification cooldown rate of 100 degrees Fahrenheit/hour. The licensee began cooling down at a rate of 200 degrees/hour. The licensee provided appropriate justification to exceed this Technical Specification limit.

At 11:25 a.m., the licensee repaired and started Diesel Generator 1 and powered safeguards equipment and began recovery from the event. The licensee entered Section 14.0, "RCS Inventory Control," of Emergency Operating Procedure EOP-20, and began to recover reactor coolant system level.

At 11:43 a.m., personnel entered containment to close Valve HCV-348. After the valve was closed, the control room was not informed that the personnel had exited containment. This appeared to be a lack of communication between the operations support center, technical support center, and the control room.

After the recovery began at 12:05 p.m., numerous suggestions were being presented to further stabilize the plant. However, these suggestions were not being presented in a organized manner. The licensed senior operator took the initiative and requested a meeting be held to present all suggestions and develop a formal plan of direction during the recovery phase.

3 TECHNICAL SUPPORT CENTER (82301-03.03)

The inspectors team observed and evaluated the technical support center staff as they performed the full range of tasks necessary to respond to the exercise scenario. These tasks included detection and classification of events; notification of federal, state, and local response agencies; analysis of plant conditions; formulation of corrective action plans; and, prioritization of mitigating actions.

Technical support center staffing and the transfer of emergency response duties were accomplished in a timely manner with minor holdups caused by a delayed pager activation. The facility was staffed with minimum staffing levels within 15 minutes of the "Alert" declaration at 7:54 a.m., and was activated at 8:14 a.m. The technical support center was declared operational, and command and control was transferred from the control room to the technical support center at 8:40 a.m.

The technical support center site director properly escalated the emergency classification from an alert to a site-area emergency based on Emergency Action Level 11.12, "Plant Conditions Warrant Activation of All Emergency Facilities." The inspectors observed the licensee notifying state and local officials of the upgrade. Emergency Notification Form FC-1188 was properly completed with applicable information and had the approval of the site director. Emergency action levels were continuously reviewed by the technical support center director, site director, and the assistant site director.

Security personnel performed well. Evacuation of non-essential personnel was conducted at the alert level and accountability was completed within 29 minutes; however, this information was not relayed to the technical support center director until he requested an update at 8:46 a.m. The technical support center security coordinator recognized degrading plant conditions and plume location, and recommended that the radwaste building compensatory guard watch be secured due to exposure concerns and that a camera be used to monitor the required locations. This recommendation was accepted and the security guard was removed from the plume location.

Communications within the technical support center were satisfactory. The inspectors observed numerous examples of open-ended communications (i.e., no repeat-backs). No problems were identified during these communication lapses; however, the inspectors identified this as an area for improvement. Other examples of communication weaknesses included infrequent facility announcements, confusion over the technical support center operational status with regard to assuming command and control, and the failure of technical support center staff to communicate a potential ground level release path through the damaged door between the auxiliary and radwaste buildings and the broken windows of the radwaste building.

Another improvement area identified concerned the updating of equipment priorities and the equipment priority status board. The inspectors noted that after offsite power was lost at 10:06 a.m., from downed power lines, no effort was made to repair Diesel Generator 2 until the recovery phase of the event at 11:50 a.m. Therefore, for over 1.5 hours, with no emergency makeup available, primary leakage, and core uncovery approaching, Diesel Generator 2 was neglected and no repairs were attempted. However, during this time, the licensee was attempting to repair Diesel Generator 1 and exploring contingencies to get water into the reactor using the emergency diesel feedwater and firewater pumps. No feasible emergency makeup water flow paths were identified using the diesel powered pumps; therefore, the licensee limited themselves to one success path to mitigate the accident, Diesel Generator 1. Diesel Generator 1 was successfully repaired and the makeup water to the reactor was restored.

On numerous occasions, the inspectors noted a long lag time between the setting of new equipment repair priorities and the updating of the status board as well as the failure to update the equipment board based on changing plant conditions. For example, a seismic event occurred at 9:00 a.m. and a reactor trip at 9:01 a.m. followed by several component/equipment failures.

The inspectors determined through a review of the technical support center log that the technical support center site director did establish new repair priorities at 9:21 a.m. and did convey these to the operations support center; however, no announcements or updates to the priority status board were made in the technical support center until later. The equipment priority board was not updated until 9:36 a.m.

Overall, technical assessment of plant conditions and contingency measures was excellent. Engineering personnel and the reactor safety coordinator were continua.ly looking ahead and formulating contingencies and recommendations based on known plant conditions. Examples of proactive recommendations and actions included refilling the reactor coolant system using alternate sources, depressurizing the reactor to minimize leakage outside containment, and evaluating the feasibility of manually shutting a motor-operated valve with a high differential pressure across the valve seat to isolate the reactor coolant system leak.

During the accident, problems with predicating core uncovery times were identified at the technical support center. Procedure EPIP-TSC-8, "Core Damage Assessment," Revision 8, provides guidance on performing core damage assessments and also includes core uncovery time predictions. Core uncovery times can be used to assess plant conditions, to formulate corrective actions, and to make protective action recommendations based on the real or potential loss of the third fission product barrier (i.e., failure of the fuel cladding).

The inspectors observed tha. the reactor engineer performing the predicated core uncovery times was unsure of how to use Attachment 6.4, "RCS Steaming Rate vs. Time after Trip," and requested help from the reactor safety coordinator. The reactor safety coordinator did not have the time to assist and told the engineer to contact another engineer for assistance. For approximately 1 hour, the inspectors observed the reactor engineer unsuccessfully attempt to calculate core uncovery times. Nonetheless, the inspectors did note that predicated core uncovery time of approximately 1 hour was given to the technical support center director. During followup conversations, the reactor safety coordinator indicated that this time was based on "best guess" estimates and not from data collected and analyzed in accordance with Procedure EPIP-TSC-8. During the post-event reviews, the reactor engineer stated that he had never performed core uncovery predications and was having difficulties predicting uncovery times. Procedure EPIP-TSC-8, Attachment 6.1, "Core Damage Assessment Summary," and Attachment 6.5, "RCS Inventory vs. Time Plot," were not completed as required by the procedure. Based upon the inspectors' observations, review of exercise paperwork, and discussions with numerous licensee personnel, the inspectors determined the failure to have an individual properly trained and capable of performing core uncovery predications necessary to evaluate plant conditions was an exercise weakness (285/95015-01).

A post-drill critique was conducted in the technical support center by licensee personnel. With only minor exceptions, licensee personnel identified

similar concerns and observations to those identified by the NRC inspectors. The post-drill critique was self-critical and detailed. Licensee personnel did a very good job evaluating their own performance.

4 OPERATIONS SUPPORT CENTER (82301-03.05)

The inspectors evaluated the performance of the operations support center staff as they performed tasks in response to the exercise. These tasks included facility activation, providing support to operations, and in-plant emergency response team coordination.

The operations support center was promptly staffed and declared operational within 32 minutes following the alert declaration. The operations support center was efficiently readied for communication and for team formation and dispatch. Communication between the operations support center director and support staff, technical support center, and simulator control room personnel was very good and effective in carrying out the duties of the operations support center.

In-plant emergency teams were coordinated in response to requests from the technical support center and control room simulator. Emergency team players were provided excellent briefings prior to dispatch into the plant. Members of the ten teams dispatched from the operations support center were briefed on assigned tasks, appropriate routes, tools, potential hazards, communications methods, and radiological concerns. Debriefing of the in-plant emergency teams was done immediately after they returned to the operations support center. The debriefings provided an adequate assessment of the assigned task and plant conditions. Communication between the operations support center and the in-plant teams was frequent and effective. Radiological conditions and the progress of repair efforts were promptly reported to the operations support center.

The level of simulation of several activities during the exercise precluded the evaluation of individual knowledge of duties. The following are examples of simulation problems that were discussed with licensee representatives. First, members of the Alpha team (assigned to shut Valve HCV-348) did not demonstrate scenario Participant Guideline 16, which states, "Appropriate on-site participants will demonstrate the procedure for donning SCBA's once but will not discharge air tanks."

Members of the Alpha team, who were issued self-contained breathing apparatuses as instructed by their radiological work permit, did not demonstrate the procedure for donning them. In fact, the only time a self-contained breathing apparatus kit was opened was at the request of an NRC inspector to check the calibration date and the material condition of the apparatus.

Second, in-plant team players were allowed to simulate equipment and supplies that were actually available for their use. This was contrary to the Conduct of Exercise, Controller/Evaluator Guidelines, listed in Volume 1, Emergency Preparedness Exercise Manual, which stated, "DO NOT: Allow participants to simulate equipment or supplies if these are actually available for their use, i.e. a dose rate instrument that malfunctions in the field should be replaced, don't let the participant simulate obtaining a replacement or ignore the malfunction."

Examples of failing to adhere to this guideline included:

- Alpha team members did not carry flashlights with them as they were going into containment to repair Valve HCV-348. During this time the site was in station blackout.

- No members of the in-plant teams demonstrated the procedure for donning protective clothing.

A third example of a problem arising from simulating events was the use of the surrogate tour system by the Alpha in-plant team to simulate going into containment to repair Valve HCV-348. During this simulated event, the valve was verbally declared to be repaired by the controller even though the Alpha team members never reached Valve HCV-348 on the surrogate tour system. A final example occurred when the Alpha team members carried their lunches with them as they simulated the repairing of Valve HCV-348, which is in containment. Overall, this level of simulation was identified as an exercise weakness (285/95015-02).

Poor radiation protection practices were observed at the operations support center. This shortcoming was identified as an area of improvement. A member of the Delta team (assigned to repair the Valve HCV-348 breaker) demonstrated poor radiation protection practices by wearing the assigned Alnor dosimeter below his waste in his front pants pocket.

5 EMERGENCY OPERATIONS FACILITY (82301-03.04)

The inspectors observed the emergency operations facility staff as they performed tasks in response to the exercise. These tasks included facility activation, development and issuance of protective action recommendations, notification of State and local response agencies, dose assessment, analysis of plant conditions, and direct interactions with offsite agency response teams.

The emergency operations facility had been redesigned prior to the exercise and the new design appeared to be a substantive improvement. The facility is required to be staffed and operational within 1 hour after the declaration of a site-area emergency. The emergency operations facility had the minimum staffing for activation within 40 minutes after the alert declaration, and assumed command and control within 19 minutes of the 9:06 a.m. site-area emergency declaration. Upon arrival, assigned emergency response personnel

immediately readied the facility, obtained necessary procedures, and established communication links. The transfer of emergency director duties occurred at about 9:25 a.m. Upon their arrivals, offsite representatives from the states of Iowa and Nebraska were quickly and effectively integrated into the emergency response organization.

Command and control in the emergency operations facility were generally good. Regular briefings (every 30 minutes) were conducted to update facility group leaders. Input from functional area team leaders and state representatives was solicited during the briefings. General facility priorities were also discussed. In addition to the briefings, the operations liaison and emergency planning coordinator made periodic announcements to alert facility personnel of current conditions. Communications flow to the emergency operations facility was not fully adequate. For example, no indications were noted that the emergency operations facility was notified that the auxiliary building roll-up door were blocked open or that the windows in the radwaste building were broken. Both constituted potential unmonitored release paths.

Notifications to offsite response agencies of the general emergency and appropriate protective action recommendations were made in a timely manner and, generally, included appropriate information.

The emergency operations facility's performance in the area of protective action recommendations was good. Feedback from state representatives on the implementation of offsite protective actions was maintained current on a protective actions status board, and was briefed by the lead state representatives in the half-hour facility status update briefings.

Existing status boards in the emergency operations facility were generally kept current.

Dose assessment activities were performed satisfactorily in the emergency operations facility. By about 9:45 a.m., protective measures had determined that field team readings were not consistent with computerized dose projections based on plant conditions. This prompted confusion about where, or how, the releases may have occurred. Coordination with the state elements and the technical support center was generally good. Several dose-projection scenarios were developed, and corresponding offsite doses were promptly calculated via the dose-assessment computer (EAGLE). Meteorological conditions were closely monitored. Habitability of the emergency operations facility was properly considered during the exercise.

Interactions with offsite officials, a mock NRC site team, and other organizations were excellent. Offsite officials were given separate briefings and invited to provide input or ask questions during utility briefings. Collocation of the NRC site team and utility representatives appeared effective. One item identified by the licensee as a weakness was also identified by the NRC inspectors as an area for improvement. At 12:06 p.m., the operations liaison indicated that Fan VA-40C, an auxiliary building exhaust fan, was on and running. This was identified at that time by

protective measures as a release path; however, it's operation had not been briefed to offsite representatives prior to being turned on. State representatives indicated that they would have desired pre-knowledge of such a release for evaluation and possible protective actions planning.

6 SCENARIO AND EXERCISE CONDUCT (82301)

The inspectors made observations during the exercise to assess the challenge and realism of the scenario and to evaluate the conduct of the exercise.

A staggered shift change was conducted by the licensee of staffs at the technical support center, the operations support center, and the emergency operations facility. This change was effective and determined to be an exercise strength.

The inspectors determined that the scenario was sufficiently challenging to test the licensee's emergency response capabilities and demonstrate agreed upon on-site exercise objectives. Exercise control was adequately maintained by controllers with the exception of the simulation of activities (see Section 4 above).

7 LICENSEE SELF-CRITIQUE (82301-03.13)

The inspectors observed and evaluated the licensee's post-exercise facility critiques and the formal management critique on November 16, 1995, to determine whether their process would identify and characterize weak or deficient areas in need of corrective action.

The inspectors determined that the post-exercise critique process was effective and provided good input into the management review system. The findings included strengths, weaknesses, and observations. In the management critique, the licensee identified five weaknesses or items warranting corrective action: (1) Emergency Plan Implementing Procedure EPIP-OSC-2 map and its associated wording in regards to plant/site evacuation was confusing/incorrect; (2) the states were not consulted before starting auxiliary building fans, (3) core uncovery predictions were not available to the emergency director, (4) failure to aggressively pursue the release pathway to explain field team/dose assessment discrepancies as required by Procedure EPIP-RR-22, and (5) differences between the core damage assessment procedure and RADDOSE V. Most improvement items identified by the inspectors were also identified by the licensee.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

*R. L. Andrews, Division Manager, Nuclear Services
*J. W. Chase, Plant Manager
*O. J. Clayton, Manager, Emergency Planning
*G. Cook, Supervisor, Station Licensing
*S. K. Gambir, Division Manager, Nuclear Engineering
*J. K. Gasper, Manager, Training
*W. G. Gates, Vice President, Nuclear
*R. L. Jaworski, Manager, Station Engineering
*L. T. Kusek, Manager, Nuclear Safety Review
*E. Matzke, Station Licensing
*R. G. Meng, Emergency Preparedness Representative
*T. L. Patterson, Division Manager, Nuclear Operations
*A. W. Richard, Manager, Design Engineering Mechanical
*H. J. Sefick, Manager, Security Services
*M. A. Tesar, Manager, Corrective Actions
*D. R. Trausch, Manager, Nuclear Licensing and Industry Affairs

1.2 NRC Personnel

*W. C. Walker, Senior Resident Inspector

*Denotes those present at the exit meeting

2 EXIT MEETING

An exit meeting was conducted on November 16, 1995. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee did not express a position on the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 2

Licensee Scenario Summary and Timeline

SCENARIO EVENTS

NARRATIVE SUMMARY

This scenario is based on Diesel Generator #2 becoming inoperable due to damage by a fire, Diesel Generator #1 tripping on overspeed due to a governor failure, an earthquake causing an interfacing system LOCA and Containment failure and structural damage to station facilities. An after shock causes a loss of both 345 and 161 KV offsite power and increases the RCS leak rate. This Station Blackout condition results in a sustained loss of safety injection, the LOCA results in uncovering the reactor core and fuel damage. A release pathway to the environment is created by the interfacing system LOCA to Room 13, to the Auxiliary Building Stack (until Station Blackout), to the Radwaste Building Stack and from the Radwaste Building directly to the environment.

The scenario begins with an **ALERT** being declared due to the fire in Diesel Generator Room #2 affecting one train of ESF. An hour and half later, an earthquake causes HCV-348 to go to the intermediate position (SI-188 lifts), a crack in the piping downstream of HCV-348 in Room 13 (RCS leaks into Room 13), damages to the roll up door between the Auxiliary Building and the Radwaste Building, windows in the Radwaste Building and Maintenance Shop to break, a Potable Water Line in Room 81 to break, and a loss of condenser vacuum.

A **Site Area Emergency** will be declared based on the loss of two fission product barriers. The RCS barrier is lost due to RCS leakage being greater than 40 gpm (100 gpm total) and the Containment Barrier is lost due to a failure of a piping extension resulting in a release pathway to the environment.

Dose assessments will need to be made based on two release points the Auxiliary Building Stack and the Radwaste Building Stack).

An after shock causes a loss of the 345 and 161 KV offsite power (a station blackout condition). The station blackout results in a sustained loss of Safety Injection during LOCA conditions. The after shock increases the size of the RCS leak in the piping downstream of HCV-348 to about 970 gpm. A **General Emergency** will be declared based on an imminent core uncovery with a containment failure.

Station blackout will complicate dose assessment due to part of the release being unmonitored. No stack flow will exist for the Auxiliary Building, RM-052 and RM-062 will have no sample pumps running during the station blackout condition. The release will be from Room 13, though the Roll up door to the Radwaste Building, some will go up the Radwaste building stack (RM-043 will be over ranged), the rest will exit the Radwaste Building via the broken windows.

SCENARIO EVENTS

NARRATIVE SUMMARY

Protective Actions Recommendations will be made to the States based on Plant Conditions, General Emergency, and Field Team sample results. Core damage can be minimized by completing repairs to a Diesel Generator, ensuring makeup water is available to the SIRWT, and unjamming and shutting HCV-348 inside the containment.

A recommendation for establishment of recovery operations or termination of the Emergency Classification is made by the Command and Control position.

SCENARIO EVENTS

TIME LINE

0700 INITIAL CONDITIONS - The plant is operating in Mode 1 at 100 percent power.

0730 A small lube oil leak forms a pool of oil underneath FO-19-1A and FO-19-1B in Diesel Generator Room #2. The oil starts to burn. Fire detector FD-64-01 alarms in Diesel Generator Room 2. Map on PC-66 shows FD-64-01 in alarm (red). AI-149 in the Upper Electrical Penetration Room is in alarm and reads "RM 64 FLAME DET."

0734 Fire detectors FD-64-03, 04, 05, and 06 alarm indication on PC-66 and AI-149. Diesel Generator #2 Deluge system is actuated, causing a Fire Main Low Pressure alarm and automatic startup of fire pump.

0740 Sprinkler system extinguishes fire in DG-2. Heavy black smoke in DG-2.

0745+ ALERT declared based on EAL 6.2 "Fire Affecting One Train of ESF."

0750+ Fire is out in Diesel Generator Room #2.

0750+ Damage inspection results: Room is full of smoke, water is all over the DG#2 room, lube oil is dripping from a fitting, the inlet hoses to FO-19-2A and FO-19-2B are melted and deformed, and all four spin-on filters are warped.

0900 Earthquake is felt in the plant at all facilities at the Fort Calhoun Station (Lasts about 40 seconds). "STRONG MOTION SEISMIC EVENT IN PROGRESS" alarm is actuated.

0901 CAS receives an alarm on the roll up door between Auxiliary Building and the Radwaste Building (BAST area). Its door has been damaged and is hanging ajar out of it tracks.

0901 HCV-348 indicates intermediate, causing the relief valve SI-188 cycle open and shut relieving to the Pressurizer Quench Tank (about 76 gpm). A 24 gpm RCS leak begins in a small crack in the piping on the containment side of HCV-347 in Room 13. The total RCS leakage is about 100 gpm. Area radiation monitors and the PINGs in Corridor 4 start to raise, and RM-052 and 062 increase. Room 13 is humid and steam can be seen coming from the piping near HCV-347.

SCENARIO EVENTS

TIME LINE

0901+ Diesel Generator #1 Trips on overspeed when started (DG-1 will most likely start on a manually initiated reactor trip following a loss of condenser vacuum). Diesel Generator #1, investigation results in finding the oil line broken on the inlet to the Governor Booster pumps. The oil level in the Governor is empty. The diesel was started with the fuel racks in the full fuel position (normal) with the governor inoperative due to the loss of oil.

0915+ Site Area Emergency declared based on EAL 1.16 "Failure/Challenge to two Fission Product Barriers." RCS barrier is lost due to RCS leakage being greater than 40 gpm. The Containment barrier is lost due to failure of the shutdown cooling suction line cracking, creating a release pathway to the environment.

0915 Potable water line downstream of LCV-1650 (Potable Water Tank PW-1 Level Control Valve) in Room 81 cracks. Potable water leaks out onto the Room 81 Floor at about 100 gpm. Potable water line pressure is about 38 psig. On PI-1601-1 (Water Plant Area) and PI-1601-2 (Water Plant Control Room). Potable water pressure will return to normal pressure 125 psig after the leak is isolated.

0920+ Rupture Disk ruptures on Pressurizer Quench Tank, Containment Area monitors rise.

0940 Loss of condenser vacuum causing a turbine and reactor trip.

1005 A second Earthquake is felt at all facilities at Fort Calhoun Station (lasts about 25 seconds).

1006 Loss of 345 and 161 KV offsite power. Security officer reports that the 345 KV tower just northeast of the protected area has fallen and that 161 KV lines are on the ground. (13.8 power remains to the Warehouse, TSC, Maintenance Shops, C & RP Building, Administration Building, and the Radwaste Building. 13.8 KV is also available to T1B-3C-1B3C transformer).

1006+ System Operations calls the Control Room and informs them they will investigate the Loss of 345 and 161 power.

1010 The RCS leak rate increases through HCV-347 and the crack in the piping near HCV-348 to 970 gpm.

1015+ General Emergency based on EAL 1.19 "Imminent Core Uncovery With Containment Failure or Challenge." Adequate safety injection flow cannot be maintained and a containment failure exists.

SCENARIO EVENTS
TIME LINE

1110+ Core uncovery; core damage begins.

1130+ DG-1 or DG-2 repaired.

1145+ Repair Team unjams HCV-348. HCV-348 is shut.

1200+ Recovery Phase.

1300 Secure from the exercise.