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Facility Name: Oconee Nuclear Station Units 1, 2, and 3

Inspection Conducted: March 12-15, 1991

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I. INTRODUCTION - AIT FORMATION AND INITIATION

A. Background

Oconee Units 1, 2, and 3 are B&W pressurized water reactors with steel lined prestressed post tensioned concrete cylinders with hemispherical dome containments. The units are located 8 miles north of Seneca, SC in Oconee County, SC. Unit 3 went critical September 1974 and was commercially operational in December 1974. Units 1 and 2 went critical in April and November 1973 and were commercially operational in July 1973 and September 1974 respectively.

On Friday March 9, 1991, at 11:40 a.m., the licensee notified the NRC Headquarters duty officer of the following event:

> "Unit 3 experienced an 18 minutes loss of RHR following an inadvertent draining of the reactor vessel (RV) during testing. During the performance of VOTES (patented process similar to MOVATS) testing of the reactor building emergency sump suction valve (3LP-19) the electrician cycle. the valve to check the limit switch position without notifying the control room operators as previously agreed to. The valve was left open establishing a drain path from the RV to the reactor building emergency sump. Reactor vessel water level (RVWL) decreased from the RV head flange (80" indicated) to bottom of the hot leg RV penetration (-18" below indication). At 0848 EST, control room operators received a normal sump high level alarm (15") and observed a loss of RVWL. At 0853 EST the operating LPCI pump "A" was manually secured after cavitation was observed. Operators isolated the drain path by closing 3LP-19 and returned RVWL up to RV head flange using gravity flow from the borated water storage tank. LPCI pump "A" was vented and returned to service at 0911 EST thereby restoring RHR decay heat removal. Primary temp as measured on the discharge side of the LPCI pump had increased 5F from 94F to 99F during this 18 minutes. Unit 3 has been shut down for 19 days with a core reload of 1/3 new and 2/3 burned fuel. During this transient reactor protection (RP) techs directed a precautionary evacuation of the reactor building. Only one worker was reported in the vicinity of the core (working overhead on the polar crane). RP tech reported to the control room that the maximum rad reading measured was 8 REM above the RV (no normal level readings were available to the individual making the report and that the maximum personnel dose was 15 mrem. However this also included exposure during the period prior to the inadvertent draining. RP techs restored access following surveys. Licensee was unable to provide time when testing which established the drain path occurred. Approx 24,000 gallons of borated water was drained to the sump through an 8" line. Water height above active fuel was requested but not provided. Licensee is performing cleanup operations at this time and has cause under investigation."

The licensee's investigation into this event determined that a blank flange was installed on 3LP-20 instead of 3LP-19. This was due to a labeling error and miscommunications between the workers installing the flange and their supervisor.

B. AIT Formation

On the morning of Monday, March 11, 1991, the Regional Administrator, after further briefing by the regional and resident staff and consultation with senior NRC management, directed the formation of an AIT from Region II, AEOD and NRR personnel. The AIT was to be headed by a Region II Reactor Projects Section Chief. The basis for the formation of the AIT was to gain a clearer understanding of an event related to the generic concern of shutdown risk management.

C. AIT Charter - Inspection Initiation

The Charter for the AIT was prepared on March 11, 1991. The special inspection commenced with an Entrance Meeting and licensee management briefing at 9:30 a.m. on March 12, 1991. The Charter for the AIT specified that the following tasks be completed:

1. Develop and validate the sequence of events associated with the March 8, 1991, Loss of Decay Heat removal at Oconee. This sequence should begin with plant conditions immediately prior to the event, including known significant deficiencies in safety-related and balance of plant equipment, and extend until the plant was stable on the Decay Heat Removal System.

2. Evaluate the significance of the event with regard to radiological consequences, safety system performance, and plant proximity to safety limits as defined in the Technical Specifications.

3. Identify procedures that were in place to generally avoid perturbations to the RCS and/or systems necessary to maintain the RCS in a stable and controlled condition during reduced RCS inventory conditions. Also, evaluate the effectiveness of those procedures as compared to the recommendations of GL 88-17.

4. Evaluate the degree to which prior work planning for the outage could have precluded this event. Also, evaluate the adequacy of administrative controls and implementation of these controls in relation to the misplaced flange, incorrect hydro and resulting loss of coolant path upon cycling of the emergency sump valve. (Include the following aspects: (a) Independent Verifications, (b) Verbal Communications, (c) Outage Control, and (d) System Engineer Involvement).

5. Determine if the licensee had procedures to control shutdown risk during the outage by coordinating equipment availability with vulnerable plant conditions such as reduced reactor coolant system inventory. Also, determine if the licensees' actions took into consideration the findings in the Vogtle IIT Report (NUREG-1440 and IN 90-25).

6. Evaluate the accuracy, timeliness, and effectiveness with which information on this event was reported to the NRC. Also, evaluate the adequacy of the event classification.

7. Identify any human factors, training or procedural deficiencies related to this event. Specifically, evaluate the effectiveness of the procedure for recovery from loss of RHR (Decay Heat Removal) which was used during this event.

8. Determine if any of the following played a significant role in the event: plant material condition; the quality of maintenance; or the responsiveness of engineering to identified problems.

9. Evaluate operator action during the Unit 3 event of March 8, 1991, and subsequent equipment recovery.

10. Evaluate management involvement during the Unit 3 event and the subsequent recovery.

11. For each equipment malfunction or personnel error to the extent practical, determine:

    a. Root cause.

    b. If the equipment was known to be deficient prior to the event.

    c. If equipment history would indicate that the equipment had either been historically unreliable or if maintenance or modifications had been recently performed.

    d. Pre-event status of surveillance, testing (e.g., Section XI), and/or preventive maintenance.

    e. The extent to which the equipment was covered by existing corrective action programs and the implication of the failures with respect to program effectiveness.

12. Provide a Preliminary Notification update upon initiation and conclusion of the inspection.

13. Prepare a special inspection report documenting the results of the above activities within 30 days of inspection completion.

D. Persons Contacted

See Appendix 1

E. Acronyms

See Appendix 2

F. Documents Reviewed

See Appendix 3

II. Event Description

A. Event Overview for Oconee Unit 3

At approximately 9:00 A.M., on March 8, 1991, Oconee Unit 3 lost the DHR system due to cavitation of the operating LPI pump caused by a rapid primary system water loss. Approximately 9,750 gallons of water were drained from the RCS into containment. Another 4,500 gallons were drained from the BWST into containment for a total of approximately 14,000 gallons. The operators took prompt action to stop the water loss, refill the primary system from the BWST to allow for LPI pump operation and subsequently restarted the pump. Operation without cooling flow lasted for approximately 18 minutes.

B. Initial Conditions

The unit was in day 24 of a refueling outage. The fuel had been reloaded into the core and the transfer canal had been drained. Preparations were being made to lower the reactor vessel level to 64 inches for installing the upper core internals. Primary system temperature was 94 degrees as indicated on the LPI pump suction. Reactor vessel level was at 76 inches as read on LT-5, reactor vessel level instrument. The 3A LPI pump was in operation in the DHR mode. The 3C pump and train B were available at the time (See Figure 1).

At 7:30 a.m., on March 8, the shift I&E technicians requested authorization to perform testing on valve 3LP-19, emergency sump suction valve. The line from the sump to this valve was thought to have had a blank flange installed. In reality, the blank flange had erroneously been installed on 3LP-20. The operations personnel authorized the activity but requested that I&E notify the CRO prior to operating the valve. The valve had previously had its power supply breaker racked out to preclude inadvertent operation.

C. Detailed Sequence of Events as Verified By the AIT and Licensee Personnel

DATE/TIME	EVENT
2/22/91	A maintenance supervisor instructed maintenance technicians to install a blank flange (cap) on the emergency sump suction line piping for 3LP-19 using WR 57039A.
2/22/91	Maintenance technicians questioned the maintenance supervisor concerning 3LP-19 suction piping location. The maintenance supervisor used flow diagram OFD-102A-3.1, to determine the emergency sump LPI suction line's physical layout. The maintenance supervisor told the technicians that the 3LP-19 line was the west (left) pipe. (See Figure 2)
2/22/91	Maintenance technicians entered the RB, then identified and verified the 3LP-19 suction line location per MP/0/A/1800/105, using the maintenance supervisor's directions and a handwritten indication on the RB wall. The marking on the RB wall was in indelible ink and indicated that 3LP-20 was 3LP-19. The maintenance technicians installed the flange on the west (left) line that was identified as 3LP-19 suction line by the marking and by their supervisor. This was actually the 3LP-20 suction line.
2/23/91	Operations initiated PT/3/A/203/04, Enclosure 13.2. This procedure was to be used to measure leakage from the 3LP-19, line from the reactor building. This test is required by Technical Specification 4.5.4.2. The CRO verified that the 3A emergency sump line (to 3LP-19) was properly flanged. There was no documented NLO verification. The RO certified that no flange was installed on the 3B emergency sump line (to 3LP-20) by sending a NLO to locally verify that the line did not have a flange. The NLO verified the flange position using the indelible ink markings on RB wall. Subsequent investigation revealed that the 3LP-20 upstream piping was pressurized from the containment sump side of the penetration; however, the cubicle drain from 3LP-19 was monitored for leakage on the RB side of the penetration. The 3LP-19 upstream piping was not pressurized.
2/25/91	PT/3/A/203/04, Enclosure 13.2 was completed up to but not including the step required to cycle 3LP-19.

2/25/91 - Valve 3LP-19 was repacked using Work Request 57933D.
3/5/91 This work required motor operator removal.

3/5/91 The motor operator for 3LP-19 was reconnected to the valve.

3/8/91

07:30 I&E technicians requested permission from the CRS to stroke 3LP-19. The CRS was reluctant to stroke the valve but conferred with an operations engineer. After weighing various factors, it was determined that 3LP-19 could be stroked. From discussions with the participants in this meeting, the following meeting results were obtained:

The valve stroke should be as quick as possible.
LPI pump A would be secured prior to the stroke to prevent air binding.

A communication problem occurred. Operations personnel expected I&E technicians to contact the CR prior to stroking the valve. I&E technicians did not understand that they needed to contact the CR.

07:54 Operations cleared the red tags and white tags from the 3LP-19 breaker and handwheel.

08:00 Operations racked in and closed the breaker to 3LP-19. The valve position showed closed in the control room.

08:30 An I&E technician was sent as a "human" red tag to the breaker for 3LP-19 in the electrical equipment room.

08:30 The CRO noticed a loss of 3LP-19 position indication and sent a NLO to investigate. The NLO found the I&E technician at the breaker with the breaker opened and reported this to the CR.

08:48 An I&E technician began to manually open 3LP-19. This evolution took approximately 7 minutes.

08:48 The CROs received a RBNS high level alarm annunciator. (At this time low pressure service water was being drained from the reactor coolant pump motor cooler and was flowing into the RBNS. Level was approaching the alarm setpoint. The level alarm was initially perceived as a normal alarm for the existing plant conditions).

08:48 A CRO noticed LT-5 (See Figure 3), reactor vessel level instrument, was at 20 inches and decreasing. This indicated the primary system had an excessive leak or the instrument had failed. The CROs on shift discussed the possible causes of the decreasing level. This instrument had failed in the past.

08:48:37 Emergency sump high level alarm was received in the main control room.

08:50:57 The reactor vessel ultrasonic level alarm was received.

08:50:04 The RBNS high level alarm was received.

08:52 - 08:52 The RBNS level was off scale high, the RB Emergency sump level was increasing. The operators immediately entered Abnormal Procedure AP/3/A/1700/07 and directed RP personnel at the RB personnel hatch to investigate leakage in the RB.

08:52:32 The CRO stopped 3A LPI pump due to a spike downward in amperes indicating cavitation.

08:53:58 The operators began pumping the RBNS to high activity waste tank.

08:54:51 3LP-19 reached the full open position.

08:56:30 The NLO closed the breakers for 3LP-21 and 3LP-22 and the CRO opened 3LP-21 and 3LP-22 (BWST Supply to LPI Suction). The valves were cycled open and the operators determined that 3LP-19 opening was probably the cause of the inventory loss.

08:57:21 3LP-21 and 3LP-22 were re-closed and NLOs were sent to close the 3LP-19 breaker to allow the valve to be closed to minimize the water introduction into the RB basement.

08:57:32 An I&E technician closed 3LP-19 breaker and then closed the valve from the breaker compartment.

08:58 RP personnel were notified to evacuate personnel from the basement and the third and fourth floor of the reactor building.

08:58 3LP-19 reached the fully closed position.

08:59 The I&E technician re-opened 3LP-19 from the breaker as part of his test process.

08:59:54 3LP-19 reached the fully opened position.

09:00:06 The I&E technician again closed 3LP-19 from the breaker prior to the NLOs dispatched from the control room reaching the breaker. At this time the I&E technicians were informed of the loss of inventory problem.

09:01 3LP-19 reached the fully closed position.

09:02:52 Operations then opened 3LP-21 and 3LP-22 to add water from the BWST to the reactor coolant system.

09:04:06 As reactor coolant system level recovered, 3LP-21 was closed.

09:05:32 3LP-22 was closed when vessel level reached approximately 76 inches on LT-5.

09:09 Operations vented the 3A LPI pump.

09:11:02 3A LPI pump was returned to service and DHR was reestablished. AP/3/A/1700/07 was exited. Temperature as indicated on the LPI pump suction indicated 99 degrees. The BWST level had changed by 1.7 feet (14,000 gallons).

From the sequence of events, the 3A LPI pump was shut off from 08:52:32 until 09:11:02, and core cooling was not available from 08:52:32 until 09:02:52 when the reactor vessel level recovery was initiated. The maximum temperature rise was less than 25 degrees by calculation, see paragraph VI.

III. Radiation Protection

A. Reactor Building Conditions

At 8:45 a.m., on March 8, 1991, licensee personnel were performing various maintenance activities on different elevations in the RB. There were no personnel working in the basement or in the refueling cavity areas. Several employees entered the RB at 8:47 a.m. to remove stud hole caps on the reactor vessel in the refueling cavity. The employees were given DADs for the task and entered the RB refueling floor which is located on the third floor in the RB. The workers were waiting for the HPT, assigned to monitor their job, to join them and were making preparations to enter the cavity when a vendor employee on the refueling floor noticed that the reactor vessel water level was dropping quickly. The vendor pointed this out to the maintenance crew. The DADs worn by the maintenance crew also indicated that the dose rates had increased. The vendor and the work crew decided to evacuate the area and proceeded to exit the RB. On their way out, the workers met the HPT that was to cover their work assignment. They reported their observations to the HPT and they all exited the RB and reported the changing conditions to the LHPT.

B. Health Physics Response

After reporting to the LHPT, the HPT reentered the RB to determine the dose rates near the refueling cavity. The HPT found the dose rates on the refueling floor adjacent to the reactor cavity had increased significantly. Dose rates at the south handrail of the refuel cavity had been 20-30 mrem/hr. The HPT found they had increased to 220 mrem/hr. The dose rates along the east side of the cavity that had been 70 to 80 mrem/hr had increased to 8,000 mrem/hr. The HPT reported his findings to the LHPT and reentered the RB to look for any additional personnel that may have been on the third and fourth RB elevations. There were no additional personnel on the refueling floor.

The LHPT received a call from, Unit 3 CR personnel about 8:53 a.m. reporting problems with reactor vessel water levels which was about 5 minutes after the maintenance personnel on the refueling floor had exited the RB. The CROs requested that RP personnel investigate the leakage in the RB. A HPT was sent to investigate. The LHPT received another call from a worker leaving the B Cavity stating that there was water on the basement floor that was rising fast and was about 6 - 12 inches deep at the emergency sump. About one minute later, the LHPT called the CR to determine the source of water to the emergency sump. The CR reported that the water was coming from the emergency sump suction line. A HPT roped off the primary access routes to the basement and posted the area as", Airborne Radioactivity Area, No Entry". The technician determined that there were no personnel on the basement floor and established radiation protection boundaries to the RB basement by 8:55 a.m.

The LHPT told the AIT that the containment HPTs had evacuated the third and fourth elevations of the RB and that there were no personnel on the basement floor when the spill occurred. The LHPT also reported that he had been able to communicate with the HPTs working in the RB and told them to inform other maintenance workers of the radiological hazards on the basement, third and fourth floor. The LHPT had 5 HPTs in the RB when the event occurred and he reported to the AIT that there were additional HPTs at the personnel hatch that could have been used if needed. The LHPT's foreman and supervisor were at the personnel access hatch when the event occurred and supported the LHPT's decisions.

C. Personnel Radiological Exposures

1. Contamination

Since there were no personnel working on the basement level when the event occurred and the HPTs responded quickly to control access, there were no personnel contamination events resulting from the spill. The radioactivity of the water spilled was 2 E-2 microcuries per milliliter.

2. Airborne Radioactivity

The licensee did not have a problem with airborne radioactivity from the reactor vessel drain down or from the water in the RB basement. The basement was posted as an airborne radioactivity area for control purposes when the floor was wet until air samples could be obtained and analyzed, and to limit personnel access. The AIT determined that the licensee did not have any airborne radioactivity during the event or during subsequent decontamination of the basement floor.

3. External Radiation Exposure

Dose rates on the refueling floor increased during the reactor vessel drain down. However, only a few people were in the area when the dose rates increased from 20 to 30 mrem/hr to 220 mrem/hr and they were only there for two or three minutes. The total time from start of vessel drain to restored water level was 18 minutes. The highest exposure of any worker exiting containment during that period was 40 mrem/hr.

D. Recovery

The radiation levels on the refuel floor began decreasing when the water level in the vessel began to increase. When the water level was restored, the radiation levels were back to the level that existed prior to the event.

The flow of water to the emergency sump from the RCS ended at 9:01 a.m. The water from the normal sump was drained to the waste hold-up tank at 8:54 a.m. and the water in the emergency sump was drained to the high activity waste tank later that day. The general contamination levels on the floor after drain down were approximately 500,000 disintegrations per minute per 100 square centimeters.

A licensee decontamination team began decontamination at 10:00 p.m. the day of the spill. The licensee's decontamination teams completed initial decontamination on March 9, 1991, and routine access to the basement was restored. The licensee was then able to resume routine outage activities on the basement floor. The total collective dose for decontamination to 15,000 disintegrations per minute per one hundred square centimeters was estimated to be 0.2 person-rem and took 36 man-hours.

E. Evaluation of Health Physics Response

Licensee representatives were asked by the AIT if they had considered RB evacuation. The LHPT reported that he knew where everyone was during the event and did not believe that there was a need to evacuate personnel that were not in the affected areas (basement, refuel and fourth floor). However, the LHPT reported to the AIT that he wondered why the CROs had not initiated the RB evacuation alarm with their knowledge of the low water level in the reactor vessel.

The AIT determined that the licensee did not have any special radiological emergency response procedures or training for in-plant radiological emergencies. The licensee emergency procedure that the operations staff used, AP/3/A/1700/07, during the event did have a requirement to initiate the RB evacuation alarm; but this requirement was coupled with establishing RB containment closure, which was a later step in the procedure. The AIT also determined that the CR did not have any functioning radiation monitors for the refueling area since they were being replaced during the refueling outage. All of the RB radiation monitors were out of service when the event occurred.

F. Plant Radiological Monitoring Equipment

When the event occurred the licensee did not have any remote ARMs in service. The licensee was in the process of upgrading plant monitoring capability with a state-of-the-art digital radiation monitoring system. The modification included the installation of new detectors and a computerized control and data acquisition system. The licensee's design change included:

Replacing two ARM detectors, 3RIA-2 and 3, on the main and auxiliary refuel bridges with local ARMs that have not been numbered yet. These detectors would not send a signal to the CR,

Installing a new detector, 3RIA-3, to the side of the refueling cavity walls that will send a signal to the CR,

Moving the RB personnel hatch monitor, 3RIA-4, approximately 20 feet from the previous position,

Replacing the old detector, 3RIA-5, on the incore instrument tank monitor with a new detector.

The licensee considered taking the detectors out of service one at a time, however, all of the detectors were connected to one analog readout module in the CR. The licensee discovered that they could not split out the detectors individually. Additionally, when the refueling outage began, only one of the ARMs, 3RIA-5, was operational. The AIT determined that the ARMs on the refueling bridge had not been operational during previous outages. Portable radiation survey instruments had been utilized in their place. The licensee had removed the RB ARMs and others in Unit 3 buildings the day before the event on March 7, 1991. The following is the last calibration date and alarm set points:

Detector Number	Date Calibrated	Set-points (mrem/hr) ALERT	HIGH
3RIA-2	11/19/89	50	10
3RIA-3	01/04/87	25	10
3RIA-4	Not Known	16	2.5
3RIA-5	12/03/89	50	1.5

The AIT determined that the ARMs on the refueling bridges would have alarmed during the event had they been operational, however, operating procedures would not have required the CRO to initiate the containment building evacuation alarm. The only area radiation monitor that would have automatically caused an evacuation alarm, 3RIA-4, which was at the personnel access hatch. It would have initiated an evacuation alarm if the dose rates at the personnel hatch reached 16 millirem. However, the personnel access hatch is on a different elevation and there was no indication of increased radiation levels at this location during the event.

The AIT reviewed the licensee's design change packages and safety reviews for the ARM modification. The safety review reported that the new 3RIA-3 monitor, located on the west wall beside the refuel cavity, would be capable of operating both during fuel movement and reactor power operations. In interviews with personnel overseeing the modification, the AIT determined that the licensee planned to disconnect this detector during power operations to increase its service life. The AIT discussed with licensee management that, if those were the plans for the detector use, it appears that another safety review would be required.

The licensee was also replacing the unit vent and RB monitors and connecting them to the new system control and data acquisition system when the event occurred. The licensee had the following sampling capability when the event occurred:

A portable pump was installed on the RB vent monitor supply lines. This sampled the particulate and radio iodines from this system. The filters were analyzed daily and a gas sample was taken and analyzed every 8-hours.

Six temporary monitors were continuously sampling specific areas in containment. The filters from these monitors were being analyzed every four to six hours. The monitors were located in each cavity (A and B) where the RCP motors are located, one was in the RB basement, one was on the first floor, one was on the third floor, (refuel floor) and one was on the fourth floor.

A portable continuous air monitor was also monitoring the reactor compartment for background airborne radioactivity. This monitor has the capability to detect particulate, gaseous and iodine activity. This monitor had local alarm capability but did not alarm during the event. This monitor was located on the third floor.

The licensee did not detect any airborne radioactivity greater than Maximum Permissible Concentration limits specified in 10 CFR 20, Appendix B, Table I, Column I.

The licensee was also replacing containment high range monitors, 3RIA-57 and 3RIA-58, with a new design. These high range monitors provide the CROs with an indication of the gross gamma activity in the RB atmosphere following an accident. The monitors are safety related monitors. These monitors were not in service when the event occurred.

IV. Safety System Performance and Plant Proximity to Safety Limits as Defined in the Technical Specifications

Unit 3 had completed refueling. The reactor vessel head was removed. RCS temperature was 94 degrees Fahrenheit as read on the suction side of the 3A LPI pump. LPI pump 3A was in service. This pump was secured due to cavitation, however, LPI pump 3C and train B were available for use. Plant systems required for these conditions were functional. No TS required safety limits for these conditions were exceeded.

V. Generic Letter 88-17, Loss of Decay Heat Removal

The AIT reviewed the Selected Licensee Commitment Manual, original issue date October 1, 1989, Section 16.5.3, Reactor Coolant System - Loss of Decay Heat Removal, dated February 1991. These commitments were prepared in response to GL 88-17. These commitments are applicable when the RCS level is less than 50 inches above the RV hot leg center line. The specific procedure that implements this is OP/3/A/1103/11, Enclosure 3.7, Requirements for Reducing Reactor Vessel Level Less than 50 inches on LT-5. As stated in the initial conditions of this report (paragraph II B), the RV level was at 76 inches on LT-5, consequently, no restrictions on operations were in effect. However, a comparison with the GL requirements and plant conditions when the event occurred are contained in Figure 4.

OP/3/A/1103/11 does contain specific requirements for lowering RV level less than 50 inches. Some of these requirements include; both main feeder breakers must remain energized, establishing containment closure ability, verifying that two makeup flow paths are available to maintain RCS inventory without assistance from the LPI pumps, verifying two LPI pumps are available and can be aligned to at least one LPI header capable of injecting into the core, verifying two sources of power are available to the main feeder breakers, and testing or maintenance which may adversely affect the performance of system on components required for decay heat removal are not scheduled for the deviation of operations at less than 50 inches.

The AIT reviewed procedure PT/3/A/0601/01J which verifies the capability to maintain the main feeder busses energized by the most reliable source.

Additional, actions taken by the licensee to meet GL 88-17, include requirements for exit thermocouples redundant and diverse reactor level instrumentation, and steam generator venting.

Of particular help to the operators during this event was the redundant and diverse level instrumentation. The instrumentation includes a wide range level instrument with range from the bottom of the hot leg to the reactor vessel flange. This instrument range overlaps for part of its range with the pressurizer wide range level to verify operability and calibration. In addition, an ultrasonic level detector installed in one hot leg and one cold leg alarms when level goes below the midpoint of the hot leg. This instrumentation was operable and was used during the event to verify the loss of inventory.

Additional precautions are included in the outage schedule. One schedule reviewed by the AIT for days 30-33 of the outage lists under steam generator primary side activities, item number 95, drain RCS (loops) after refueling for the unit being in mid-loop ops procedure (RCS less than 50 inches) the following:

Mechanical Maintenance

Both steam generator upper hand holds open for vent path

Prepare to close equipment hatch in 2.5 hours, close steam generator openings in 2.5 hours and close the emergency hatch in 2.5 hours.

I&E

Verify calibration of LT-5 daily

OPS

Two sources of power for decay heat
Two means of providing makeup
Two independent RCS level indications
Two LPI pumps available
BWST level must be equal to a greater than 45 feet

The licensee has addressed GL 88-17 recommendations.

The AIT reviewed AP/3/A/1700/07 which provides detailed instructions for establishing core cooling following a LPI system loss. The procedure adequately covered this specific event and contained appropriate instructions for a wide range of core cooling losses during decay heat removal.

VI. Design Calculations

The AIT reviewed design engineering calculations performed subsequent to this event. These calculations cover a variety of postulated events along with an attempt to quantify actual plant conditions during the event. The licensee calculated that the maximum bulk reactor coolant system temperature increase would have been less than 25 degrees Fahrenheit, the estimated time to saturation was 43 minutes, and the estimated time to uncover the core was 2.6 hours. These calculations were extremely conservative. The following are some of the conservative assumptions made:

- Perfect mixing - which assumed that all the heated water would reach saturation at the same time.
- Quasi-static Heatup - which assumed that all the heat generated went directly into the reactor coolant system and the fuel pins did not retain any heat.
- The water volume in the lower plenum, downcomer, the cold leg and hot leg were assumed to be thermally isolated from the core region.
- The Reactor Pressure Vessel and internals heat-up was neglected.
    - 102 percent power history was assumed for the entire fuel cycle.
    - The Zircalloy fuel pin spacers were not included in the cladding mass.
    - For boiling considerations no credit was taken for the 2-phase mixture in the core.
    - Water volume used in the calculation assumed all internals were installed.

VII. HUMAN PERFORMANCE IMPLICATIONS

A. Introduction

Approximately two weeks before the draining of water from the reactor vessel to the containment emergency sump, two maintenance personnel were given the task of installing a blind flange on the emergency sump suction line to valve 3LP-19. Since the procedure for installation of the flange did not address how to identify the correct line, the maintenance supervisor, based on review of a drawing, suggested that the flange be installed on the left emergency sump suction line. However, the drawing used was a schematic and not intended to provide information on true physical location. In reality, the suction line to valve 3LP-19 was the one to the right. When the maintenance personnel reached the emergency sump location, a hand written, non-standard label on the wall above the sump also designated the left line as 3LP-19. They proceeded to install the flange on the left. This was the line leading to valve 3LP-20.

Once the initial installation was made of the flange on the line to emergency sump suction valve 3LP-20, the plant was aligned so that opening 3LP-19 would result in a flow path from the reactor, through the open DHR system hot leg suction line, into the emergency sump.

Other human performance aspects which contributed to this event include: (1) additional independent verifications did not detect the error in the initial flange placement, (2) maintenance and operations personnel failed to report the reliance on a non-standard label, and (3) communications failures occurred between control room operators and maintenance personnel.

These human performance aspects raise concerns relating to administrative controls, training, and procedures as they apply to maintenance activities and operations. These concerns are not limited to shutdown conditions, but could apply to normal operation as well.

The human performance implications of this event were investigated by the AIT based on interviews of plant staff and review of documents. Those interviewed included operations staff, I&E and mechanical maintenance staff, training staff and integrated scheduling personnel. Documents reviewed included normal and emergency procedures, maintenance procedures, training procedures, plant drawings, and operator's logs. A complete list of those interviewed in this phase of the investigation and a list of the documents collected are included in Appendices A and C.

To aid in understanding of the human performance implications of this event, an EVENTS AND CAUSAL FACTORS CHART was constructed. (See Figure 5). The chart is a visual presentation of the sequence of events and the causal factors for those events. The sequence of events is along the top, from left to right and the causal factors are listed below each event. From the chart it is clear that numerous causal factors are involved in this event; no individual cause stands alone. The human factor implications section of the AIT addresses most of the causal factors identified in the chart.

B. Contributing Causes

The failure of plant maintenance personnel to install the flange on the correct emergency sump suction line as part of a LPI system leakage surveillance procedure resulted in the discharge of reactor water to the RB emergency sump. Defense in depth is important to maintaining safe operation of nuclear plants. Contributing causes are factors which reduce the level of protection by reducing the defense in depth. Human errors are seldom the result of a single "root cause." The following discussion of contributing causes ranks these causes in approximate priority of their importance as the AIT judged them.

1. Labeling

The maintenance crew was instructed to install the flange on the line which connected to valve 3LP-19. Non-standard labeling of the pipe ends in the emergency sump (wall markings) directed the maintenance crew to the wrong line. The licensee could not provid information on how or when the non-standard label occurred. The pipe penetration was labeled, but that information was not included in the installation procedure.

The likelihood of error was much greater due to the incorrect non-standard labeling. No one who observed the non-standard label questioned the appropriateness of relying on it for proper identification. The station procedure on labeling, OMP 4.5, focuses on labeling of components but does not address labeling of flange locations. The plant program for labeling of components is the responsibility of the operations department. Over the past few years, the labeling in the plant has been greatly improved. However, the licensee does not consider a pipe to be a component, so the need for a standard label to identify the pipe was not considered, even though proper identification was important to the correct performance of regularly scheduled surveillance activities.

2. Independent Verification

Independent verification of the flange placement was required by the procedures for initial installation, MP/0/A/1800/105, Steps 6.5 and 11.2.1, and for subsequent performance of pressure test, PT/3/A/203/04, Enclosure 13.2, Step 1.1 and 1.2, of the line section. The verification process in the first instance was not adequate in that the two maintenance technicians who selected the pipe on which to install the flange acted in parallel rather than independently. Later, when preparing for pressure testing, the initial verification step was signed off by the CRO based on a verbal statement from the maintenance technician that they had completed the work request rather than sending an operations person to look; and the second verification step was unsuccessful because the non-standard label and the previous placement of the flange were accepted as correct.

Station Directive 2.2.2 establishes the policy on independent verification; blank flanges are mentioned but there is no caution against using non-standard labels. Prior to this event, several NLOs and maintenance personnel concurred that the flange was installed on the proper line. However, what was actually verified was that the flange was on the pipe with the non-standard label designated "LP-19."

3. Maintenance Procedures

Neither the work request nor the procedure used for installation of the flange addressed the difficulties in identification of the proper line. Consequently, the maintenance technicians who were assigned to install the flange relied on verbal guidance from their supervisor, which was confirmed by an incorrect non-standard label. Prior to reaching the sump, they did not know what type of local identification would be available. The procedure also did not provide the penetration number.

Station Directive 3.2.1, provides guidance for proper identification of components for maintenance activities. Labels for the pipe containment penetration numbers were available at the sump location and could have been used by the plant personnel for proper identification.

4. Use Of Drawings For Component Identification

The mechanical maintenance supervisor used an OFD to identify the pipe in the emergency sump on which to install the flange. Based on the OFD, he told the maintenance crew to install it on the left side which was in error. The training organization has a Basic Procedures Lesson Plan, IE-ON-ADMIN-001, which contains information on use of OFDs, but lacks a precautionary statement on use for physical location of components. This drawing did contain relevant penetration number designations which could have been used for accurate identification of the proper flange location. These were not used. Physical layout drawings which would properly locate the lines, were available but not conveniently accessible for use by maintenance or operations plant personnel.

5. Communications

On March 8, prior to the event, the I&E maintenance technician, accompanied by a nuclear production engineer representing the operations engineering section, held a discussion with the unit CRS regarding the work to be done on valve 3LP-19. It was understood by all parties that the CROs would stop the running LPI pump prior to the technician opening the valve. This would reduce the probability of air binding the pump.

The discussion was concluded without a clear statement of the agreed position. The CRS thought that the maintenance technician would call the control room prior to opening valve 3LP-19. However, the maintenance technician thought that the pump would be tripped immediately and that he could work on the valve as long as he was done quickly. Thus, the CR was not notified when the valve was opened and, instead, became aware the loss of inventory by the actuation of alarms in the CR. Rapid response by the control room operators minimized the consequences of this particular communications shortfall. This appears to be a case where better communications, including repeat-backs, would have been effective. Also, during this event, the maintenance technician did not hear the CR calling him on the page system because the technicians was concentrating on the task he was performing.

Effective communication would not have prevented the loss of inventory from occurring. However, the operators may have responded somewhat more quickly if the technician had called the CR immediately prior to opening the valve.

C. Control Room Operator Response

The AIT team interviewed all the operators present in the CR during the event. The purpose of these interviews was to identify what happened in the CR and to evaluate operator actions during the event and subsequent recovery. The following is an annotated list of operator actions from the perspective of the operators as presented to the AIT during the interviews. This list is not meant to represent the chronology of the event but to present relevant operator actions.

Based on a 7:30 a.m. discussion held between CR personnel, a nuclear production engineer and two electricians from I&E, the SRO believed that I&E would notify the CR just prior to cycling 3LP-19 in order to allow the CR to stop LPI Pump 3A to prevent air from binding the pump.

ROs were not aware that I&E personnel were opening valve 3LP-19. One operator believed that the CR was going to stroke valve 3LP-19. Some of the operations department personnel believed that maintenance was performing a VOTES test.

Operators acknowledged the RBNS "high" level alarm actuation, however, the alarm was expected due to other ongoing outage activities. Operators acknowledged the RB emergency sump high level alarm. Operators observed LT-5 reactor level indication drop from 76 inches.

The operator questioned his supervisor with regard to possible testing on LT-5 since the instrument had failed in the past. They decided that the instrument was correctly indicating a problem.

The level drop was confirmed by the RV ultrasonic level alarm which appeared on the front panel display. The operators observed LT-5 reactor level offscale low. The operators thought that it took only 1 minute for LT-5 to drop to offscale low.

Operators had expected the flange to leak somewhat during the period that valve 3LP-19 was opened.

Operators entered abnormal procedure AP/3/A/1700/07. Operators observed a downward spike in the pump current (amps cycling) of the LPI Pump 3A (due to cavitation). Operators stopped LPI Pump 3A.

Operators opened 3LP-21 and 3LP-22 (3A LPI BWST suction and 3B LPI BWST suction) which were red tagged in attempt to restore reactor level. They were closed when it was discovered that 3LP-19 was probably open.

Since the breaker for 3LP-19 was open, there was no CR indication of the position of 3LP-19. Operators questioned whether maintenance was opening the correct valve. Operators decided that valve 3LP-19 or 3LP-20 was the source of the leak.

NLOs were sent by the CR to close 3LP-19, however, I&E technicians were already electrically shutting the valve from the breaker. Valve 3LP-19 was shut locally from the 3LP-19 breaker in order to stop the loss of water. Valves 3LP-21 and 22 were reopened to restore reactor vessel level and were closed as level recovered.

Operators vented and restarted LPI Pump 3A without any additional problems. The LPI Pump 3A had been off for 18 minutes. Operators verified operability of LPI Pump 3A.

Operations calculated that approximately 14,250 gallons of water was spilled into the reactor building basement.

Overall, it is the opinion of the AIT team that the operators performed the necessary actions to recover from the event and performed them in a timely fashion. The correct procedure was utilized by the operators. The operators were able to follow the abnormal procedure steps for loss of LPI, and when required, deviate (as when they closed 3LP-21 and 3LP-22 from the BWST shortly after opening them) as necessary to control the level.

However, communications with maintenance personnel showed some weaknesses. Prior to the event the operators were not aware of exactly what test I&E was performing. The operators were also not aware of when the testing was to take place. The operators should have been more aware of I&E's actions, in order to be able to maintain control of the position of valve 3LP-19.

D. Work Planning and Outage Control

Overall planning and scheduling of refueling outage work is the responsibility of the Superintendent of Integrated Scheduling. However, control of the actual work done on reactor systems is the responsibility of the Superintendent of Operations. Both of these individuals report directly to the Station Manager. Final determinations of detailed system alignments for systems which are "block tagged" out of service are made by the operations department. Operations is also responsible for risk management.

The following is a discussion of the planning and scheduling activities concerning the work on valve 3LP-19.

The Outage Manager's Schedule, U-3 EOC 12 Refueling Outage, is updated daily and was reissued at 8:00 a.m., on 3/8/91. This 19-page document listed major activities for a four-day period from 3/8/91 to 3/11/91 corresponding to the 24th to the 27th day of the outage. The summary of valve repair and preventive maintenance is tabulated on page 9 (9999 Valve Status Report) and states that work is in progress on 25 valves, that work was required on 771 valves during the outage, and that work was completed on 532 of the 771 valves. A section entitled "9998 Valve Work of Interest" on page 8 lists 30 primary and secondary valves including valves 3LP-19 and 3LP-20. The 3LP-19 listing is followed by the notation, "flange installed--may need MOVATS." In accord with management policy, the 3/8/91 issue of the Outage Manager's Schedule was reviewed at a morning meeting of supervisors from approximately 8:00 - 8:45 a.m.

The controlling document for work by maintenance and I&E specialists on valve 3LP-19 was WR 57933D. The clearance block on the WR form contains the acronym "OPS" for operations and the corresponding signature block initialed by an operations engineer and dated 2/25/91. The job sequence listed on the front page of the WR is:

1. Remove (Maintenance)
2. Repack (Maintenance)
3. Reinstall (Maintenance)
4. Reconnect (I&E).

Attached to the WR is a Post-Maintenance Testing Plan, form NRC 89-10, with 57933D written in the WR space and the notation "3LP-19." The tests to be performed are specified in block 3 of the form:

*Visual Inspection for Leakage--Maintenance
*Operator/Actuator Stroke Valve Diagnostic Testing-- I&E
*Stroke Time Test--Performance

Block 3 also contains columns for "Date Performed" and "Completion verified by" for the above three tests. These columns are blank, i.e., no date or initials, and the tests had not yet been performed. The operation of valve 3LP-19, both manually and electrically from the motor control center, by the I&E specialist on 3/8/91 was part of Job Sequence 4, "Reconnect-I&E," and was intended to determine that the motor operator was properly connected and that the valve position limit switches were properly set.

However, the AIT was told that the hydrostatic test had been completed on the line with the flange installed, i.e. the line to 3LP-20. This is expected since the hydrostatic connections are located on the flange which was inside containment. The leak-off path which was monitored was located on the valve which was outside containment. The technician monitoring leakage had no reason to believe he was monitoring the wrong valve. When no leak-off flow was seen at valve 3LP-19, it was assumed that the hydro was successful. Actually, the emergency sump suction line to valve 3LP-20 had been pressurized.

The WR itself contains no precautions against operation of valve 3LP-19. The OPS engineer initials and date of 2/25/91 on the front of the WR form indicated that it was possible to perform the work safely. As previously stated, the Outage Manager's Schedule of 8:00 a.m., 3/8/91, has the notation, "3LP-19 flange installed--may use MOVATS"; and since the MOVATS work requires valve operation, the implication is clear that the flange installation made it safe to open/close 3LP-19.

Cycling the valve would not have caused a problem if the flange was on the correct line. The decision of the CRO to allow the valve to be cycled was based on the knowledge that previous work orders stated that the flange had been installed on the line to 3LP-19 and a hydro had been performed on that line.

The Unit 3 CR SRO authorized the I&E specialist to operate 3LP-19 on 3/8/91 on the basis that operational safety was maintained by the installation of the flange on 3LP-19. The initial conditions for performance test PT/3/A/203/04, contained initialed verification by a CR that the flange was installed on the 3A emergency sump line to 3LP-19 on 2/23/91. This verification was in addition to the verification on

2/22/91 by maintenance specialists that this flange was installed by WR 57039A. Step 6.5 of MP/0/A/1800/105 which was attached to this WR states, "Have Operations Unit Manager specify which flange is to be installed first." The notation "LPI-19" is written in for the flange identification, not the penetration number which is "36." The name of the CR SRO is written as the "person contacted" for this identification.

Control of operational safety during the movement of 3LP-19 for the electrical re-connection was, therefore, by (1) the Outage Manager's Schedule, (2) release of the WR 57933D by an operations engineer, (3) verification both by a maintenance specialist and a CRO that the proper flange was installed, and (4) oral permission by the CR SRO to the I&E specialist to open/close Valve 3LP-19.

The follow-up testing after completion of repacking required the manual stroking of the valve to set the limit switches. Manual stroking the valve open takes approximately 7 - 8 minutes, and with the breaker open remote operation or position indication from the control room is not possible.

A review of the LPI system leakage procedure PT/3/A/203/04, showed that Change 14 (dated 4/28/89), incorporated manual operation of 3LP-19 and 3LP-20 during refueling outages to ensure that the valves would manually operate if the electrical stroke test failed. Therefore, manual operation would have been required, regardless of the fact that the repacking was performed. The only difference in not repacking the valve, would have been that the event probably would have occurred at an earlier time following installation of the flange.

The emergency sump suction line from 3LP-19 is unique among the six suction lines on the three Oconee plants in that it is the only one that cannot be isolated from the DHR suction line without interrupting the normal DHR flow path. There is no evidence to suggest that revised valve alignments in other parts of the DHR system had been used to reduce the risk during prior outages.

The AIT reviewed LP-19 and LP-20 testing for the last two fuel cycles on all three units. There was no clear pattern of which equipment was out of service or reactor vessel level. The testing had occurred at low point maintenance, at power, defueled, with the refueling canal filled and with the refueling canal drained. The work appeared to be performed when the work came due on the schedule without taking vessel level or the status of fuel in the core into consideration.

Better overall outage control could have prevented this event. Valve alignments could be made for all lines but 3LP-19 to provide double valve protection. Also, valve surveillance could be performed with the reactor defueled and drained. This had been the case during some prior outages, although there is no written guidance on the issue and the scheduling is subject to revision. In fact, in this case, replacing the valve packing delayed the tests for more that a week; by then the reactor vessel had been refueled and refilled.

E. Training and Procedures.

The AIT investigated training and procedures related to shutdown operations.

The AIT interviewed licensee personnel regarding training activities for both operations and maintenance. In the maintenance area, the issues of independent verification, identification of components, and use of drawings were discussed. It was identified that there was no written cautions on the use of schematic drawings for determining physical location or on the use of non-standard labels. Licensee discussions were corroborated by the contents of training lesson plans which were provided.

Operations training on DHR operations was also discussed. No specific simulator training on loss of inventory events had been provided. However, classroom training on the LPI procedure did address the actions to take to restore reactor coolant inventory during DHR operations.

Operator response to the event was based on knowledge and training on the LPI system. The procedures were used as a check after the initial rapid response of the operators. The AIT found that they had done as the procedure suggested. The AIT review of the procedure did not identify any problems.

Maintenance procedures for placement of the blind flange did not include guidance on identification of the correct line. The LPI leakage test procedure called for verification of the flange on 3LP-19 but also did not provide proper guidance on identification. Because of the wide range of maintenance activities, the need for directions regarding identification of a specific component should be done on a case by case basis. The maintenance supervisor indicated that, in the future, such guidance would be provided for the flange installation procedure.

F. Plant Physical Condition and Maintenance Quality

Through the interviews conducted and a plant walkdown, the AIT was able to ascertain that neither plant material condition or quality of maintenance played a role in any of the aspects of related to this event. Additionally, there was no evidence that any viable concerns related to engineering requirements of the system design were contributory to the event.

A walkdown of the unit showed that an extensive effort has been made to properly label plant components (i.e., equipment, valves, hangers, etc.) and that the general condition of the plant was good. Based on these observations and personnel interviews, it was apparent that the emphasis placed on component identification labeling had been the result of some past problems.

As stated previously, containment penetration number labels were properly affixed at the sump and shown on the OFD drawing, and had the proper reference been utilized or other verifiable means of identification used, the miss-installation of the flange may have been averted.

Similarly, a review of the maintenance WRs and interviews with several technicians and supervisors established that the quality of work performed was good. Maintenance personnel performing the work had substantial experience and sufficient training.

One of the initial causal factors of the event involved a maintenance supervisor and the use of a flow diagram for physical location of a component. The supervisor admitted that he should not have used a flow diagram. The supervisor indicated to the AIT that the Oconee piping physical layout drawings were not easily accessible.

Installation and removal of the flanges on the emergency sump lines is normally an outage task which takes only a few days. The repacking of 3LP-19 which is required every third refueling, delayed the normal schedule (between installation and removal of flange) by approximately 2 weeks.

The Oconee equivalent of the systems engineer for the LPI system is the Unit 3 Operations Manager. He was not directly involved in this event. However, a nuclear production engineer assigned to the Operations Manager for this outage became involved in discussions regarding performance of the test which was involved with the event. The maintenance technician requested permission from the nuclear production engineer to do the test. They proceeded to the CR to discuss the situation with the CRS.

G. Management Involvement During the Event and Recovery

The AIT investigated management involvement in outage planning, coordination of maintenance and operations activities, overtime issues, communications and management involvement with the event investigation and identification of root causes.

Management involvement in investigating the event began approximately two hours after the plant was placed in a safe condition. The Chairman of the SRC was responsible for overseeing the investigation. One individual on the committee was assigned the lead role for the detailed investigation of the event and the identification of the activities that led up to the event. This main investigator was on-site within two hours and interviewed all the major persons involved for the preparation of suggested corrective actions. According to the SRG charter this group is an onsite independent engineering/technical review group established to perform a function encompassing the following areas:

Review industry and inhouse Operating Experience (OE) information to maintain awareness and to incorporate into the performance of other duties. Selectively review completed evaluations of OE to independently determine the accuracy and adequacy of the evaluation and if further corrective actions should be developed.

Investigation of selected unusual events and other occurrences at each nuclear station.

Perform reviews of selected programs, procedures, and plant activities for the purpose of identifying improvements in programs, management controls, work practices, etc.

Based on SRG procedure SRG/2, the investigator is required to obtain all information and data relevant to the event using available sources. Once this information is obtained, the investigator reconstructs the event using event and casual factor charting, prepares a sequence of events, performs an evaluation and analysis, prepares conclusions, determines root cause, obtains descriptions of planned and implemented corrective actions, develops a safety analysis and documents this effort in an incident report written in the format for an LER.

The Outage Manager's Schedule was reviewed and found to be complete. It appeared to be representative of a typical Oconee outage. In addition, management holds an outage meeting every morning to review important activities of the day and discuss coordination if necessary. The outage meeting discussions did not reach the level of detail where individual valve maintenance activities were considered. This is not unusual considering the large number of activities scheduled during the period under consideration.

The Superintendent of Operations is responsible for maintaining plant configurations which assure safety during outages. Although operations personnel are routinely required to make decisions regarding plant configuration management, the AIT was not aware of any written policy guidance within the operations department with regard to maintaining defense in depth through use of double isolation or other means where possible.

The AIT checked to see that the planning and scheduling department had been appropriately involved with the WRs that were involved in this event. Each of the WRs was reviewed and the appropriate sign-offs and clearances were obtained.

Although the operations department has final responsibility for maintaining a safe plant configuration during outages, planning and scheduling decisions do have an impact on plant safety during outages. The AIT was not aware of any written policy guidance within the outage planning and scheduling organization with regard to configuration management safety considerations.

The days prior to the event were busy but not atypical for an outage. The operations shift crew was on their normal rotating 12 hour shift. There was no evidence of operators working an excessive amount of overtime. However, the electrical technician that opened the valve had been working overtime. Even though he had worked more hours than the Commission Policy Statement on Overtime allows, he had received the appropriate levels of approval from management. The recommendations of the Policy Statement were therefore not violated since the Policy Statement allows for additional overtime if approved. In fact, at the time of the event, the electrician had worked 12 days straight. He did not feel that he was fatigued, however, it is the opinion of the AIT that 12 days straight for some workers can be excessive. The licensee should review their overtime records to make sure that personnel do not work excessive amounts of overtime.

Management at Oconee was clearly interested in improving communications even prior to the event. The station management had initiated and implemented a training program for 2,500 people at Oconee including contractors. This training program included a course in "Please Listen"— which encourage employees to communicate more carefully and clearly. Part of the training emphasized "repeat backs." The licensee should investigate ways to encourage on the job use of the material covered in the training program.

VIII. Reporting

This event occurred on March 8, 1998, at 8:48 a.m., and it was reported to the NRC at 11:40 a.m. the same day. The licensee reported the event under the requirements of 10 CFR 50.72(b)(2)(iii)(B) which states that a four hour report is required for any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to removed residual heat.

The AIT reviewed RP/0/B/1000/01 which provides the guidelines for emergency classification by the licensee. Enclosure 4.1.5, Loss of Shutdown Functions, provides specific criteria for event classifications and this event was reported correctly using these criteria. The information, with minor exceptions, was accurate. The minor exceptions included that the licensee was performing MOVATS testing vice VOTES testing, the refueling outage was in the 24th day of the refueling outage vice the 19th day and the amount of water lost from the RCS and BWST was approximately 14,250 gallons vice 24,000 gallons.

IX. Conclusions

Although an extensive labeling program has been implemented at Oconee, the location for flange installation was not correctly identified and no one who saw the non-standard label reported the problem.

Several independent verifications were executed but the flange installation error was not discovered.

Because of a misunderstanding between a maintenance technician and the unit CRS, the technician opened the valve without notifying the CR.

The procedures used for installation of the flange did not include guidance for identification of the correct location.

The maintenance supervisor incorrectly used the OFD drawing to determine physical location for the flange installation.

The operators acted decisively on the loss of coolant, quickly determined the cause and returned the plant to a pre-event condition.

The calculation for determining the heat generated upon the loss of coolant was conservative and well done.

X. Exit Interview With Licensee Management

The inspection scope and findings were summarized on March 15, 1991, with those persons indicated in Appendix 1. The NRC described the areas inspected and discussed in detail the inspection results delineated in this report. No proprietary material is contained in this report. No dissenting comments were received from the licensee.

APPENDIX A - PERSONS CONTACTED

Licensee Employees

C. Baldwin, Quality Verification Manager
*B. Barron, Station Manager
R. Bowser, Scientist, Radiation Protection
D. Carpenter, Quality Assurance Inspector - Mechanical
T. Carroll, Radiation Protection Specialist
T. Curtis, Compliance Manager
J. Davis, Technical Services Superintendent
D. Deatherage, Staff Operations Manager
N. Edwards, Unit 3 Operations Manager
W. Foster, Maintenance Superintendent
D. Gordon, Operations Shift Supervisor
M. Greenwood, I&E Specialist
C. Hale, Mechanical Maintenance Supervisor
W. Holcombe, Maintenance General Supervisor
D. Howe, Nuclear Operations Specialist
*O. Kohler, Licensing Coordinator
*E. Lampe, Scientist, Radiation Protection
L. Lee, Control Room Supervisor
J. Looper, Control Room Operator
*H. Lowery, Onsite Review Group Manager
C. Matheson, Instructor, Mechanical Maintenance
J. McCall, Maintenance Specialist
K. McMurray, Nuclear Production Engineer
*R. Morgan, Quality Assurance Director
K. Owen, Instructor, I&E Training
J. Perry, Design Engineer
*S. Perry, Assistant Licensing Coordinator
W. Pursley, Associate Scientist, Radiation Protection
G. Ridgeway, Shift Operations Manager
D. Roth, Unit Supervisor
*G. Rothenberger, Superintendent of Integrated Scheduling
*S. Spear, General Supervisor, Radiation Protection
*R. Sweigart, Operations Superintendent
L. Taylor, Control Room Operator
M. Tuckman, Vice President, Nuclear Operations
V. Waldrop, Associate Instructor, Chemistry Training
J. Waldrup, Maintenance Specialist
P. Waltman, Quality Assurance Inspection Supervisor - Mechanical

G. Washbum, Nuclear Instructor Operations Training
R. Waterman, Radiation Specialist
T. Wehrman, Nuclear Production Engineer
D. White, Supervisor, Radiation Protection
J. Whitener, Nuclear Instructor, Operations Training
F. Williams, Control Room Operator
C. Yongue, Radiation protection Manager

Other licensee employees contacted included management representatives, engineers, technicians, operators, and office personnel.

NRC Representatives

*P. Skinner, Senior Resident Inspector

*Attended Exit Interview

APPENDIX B - ACRONYMS

AEOD	Office for Analysis and Evaluation of Operational Data
AIT	Augmented Inspection Team
ARM	Area Radiation Monitor
B&W	Babcock and Wilcox
BWST	Borated Water Storage
CR	Control Room
CRO	Control Room Operator
CRS	Control Room Supervisor
DAD	Digital Alarming Dosimeter
DHR	Decay Heat Removal
EOC	End of Cycle
GL	Generic Letter
HPI	High Pressure Injection
HPT	Health Physics Technicians
I&C	Instrumentation and Controls
I&E	Instrument and Electrical
IIT	Incident Investigation Team
LHPT	Lead Health Physics Technicians
LPI	Low Pressure Injection
MOVATS	Motor Operated Valve Actuation Testing
mrem/hr	Millirem/hour
NLO	Non-licensed Operator
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OFD	One Line Flow Diagram
OPS	Operations
RB	Reactor Building
RBNS	Reactor Building Normal Sump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RO	Reactor Operator
ROAB	Reactor Operations Analysis Branch
RP	Radiation Protection
RV	Reactor Vessel
SRC	Safety Review Committee
VOTES	Valve Operational Testing and Evaluation System
WR	Work Request

APPENDIX C - Documents Reviewed

A. PLANT TECHNICAL DOCUMENTS/PROCEDURES

1. Oconee Unit 3 Technical Specification, Section 4.5.4 "Low Pressure Injection System Leakage", Page 4.5-12, dated 5/30/85.
2. Oconee Nuclear Station Directive 3.2.1, Work Request, Revised 2/6/90.
3. Oconee Nuclear Station Directive 2.2.2, Independent Verification, revised 1/24/90.
4. Oconee Nuclear Station Maintenance Directive 7.5.3, Work Request Implementation, Revised 2/14/91.
5. PT/3/A/203/04, LPI System Leakage Procedure, Change 14, 4/28/89.
6. PT/3/A/203/04, Enclosure 13.2, Leakage Test of Emergency Sump Line to 3LP-19 During Unit Outage, Started 2/22/91 @ 1445 hrs.
7. PT/3/A/0610/01J, Emergency Power Switching Logic Functional Test, Changes 15 to 17, 3/12/91.
8. OP/2/A/1103/11, Draining and Nitrogen Purging of the RC System, Changes 18 to 19, 9/13/90.
9. OP/3/A/1104/04, Low Pressure Injection System, Changes 35 to 37, 11/28/89.
10. OP/3/A/1502/07 (Partial Copy), Refueling Procedure, dated 3/13/91 @ 1118 hrs.
11. Oconee Plant Commitment, Section 16.5 Reactor Coolant System, loss of Decay Heat Removal, dated 2/14/91, pages 16.5-7 and 16.5-8
12. AP/3/A/1700/07, Loss of Low Pressure Injection System, Change 2, dated 2/21/91.
13. MP/0/A/1800/105, Reactor Building Emergency Sump LPI Suction Line Flange Installation, Removal and Screen Inspection, Change 1, dated 8/1/90.
14. OMP 4-5, Station Labeling and Control Board Conventions, Revision 3, dated 7/13/90.

B. NUCLEAR STATION WORK REQUESTS

1. WR # 57039A, Rev 1, dated 2/22/91, Install and Remove 36" Diameter Cap on Emergency Sump Lines for PT/3/A/0203/04 (LPI System Leakage).
2. WR # 57933D, dated 2/25/91, Repack the Low Pressure Injection to Reactor Building Emergency Sump Isolation Line 'A' Valve LP-19.

C. PLANT TRAINING DOCUMENTS

1. IE-ON-OEP-1990, I & E Operating Experience Program, 1990 Roll-Up Training, Case Studies Concerning, Working on Wrong Component, Revision 0, dated 11/13/90.
2. IE-ON-ADMIN-001, I & E Administrative Continuing Training, Directives, Verification, Revision 0, dated 10/15/90.
3. IE-ON-OEP-1991, I & E Operating Experience Program, 1991 Roll-Up Training, NUREG 1410, Revision 0,dated 2/28/91.
4. IE-OC-OT-OCIE-004.1, I & E Station Directives/Maintenance Directives, I & E Orientation and Administrative Training, Revision 10, dated 8/8/90.

5. IE-OC-DHR-OEP, I & E Continuing Training, Case Studies on Decay Heat Removal and Refueling Activities that can lead to a Loss of Refueling Cavity Water, Revision 0, dated 11/6/89.
6. MM-OC-MOT-007, Mechanical Maintenance Orientation Training, Safety, Independent Verification, Revision 3, dated 4/6/90.
7. MM-OC-MOT-030, Mechanical Maintenance Orientation Training, Safety, Safety Tags, Revision 1, dated 4/6/90.
8. MM-OC-MOT-005, Mechanical Maintenance Orientation Training, Maintenance Management, Maintenance Work Request, Revision 6, dated 11/5/90.
9. MM-TC-FSS-SAF-06, Mechanical Maintenance, Nuclear, Fundamental Shop Skills, Safety, Independent Verification and Verification of Component Isolation, Revision 2, dated 5/22/89.
10. OP-OC-CP-015, Operator Training, Controlling Procedures, RCS Fill, Vent, and Drain, Revision 2, dated 8/30/90.
11. Traning Package 89-1, ONS Operations Response to Generic Letter 88-17 (Loss of DHR), dated 1/8/89, 27 pages.
12. OEP Review Response Form on NUREG 1410 for I & E, dated 2/5/91, 2 pages.
13. OEP Review Response Form on NUREG 1410 for Mechanical Technicians, dated 1/30/91, 1 page.
14. OP-OC-PNS-LPI, Operations Training, Primary Nuclear Systems, Low Pressure Injection System, Revision 5, dated 10/1/90.
15. Oconee Simulator Training Schedule, Class PTRQ-1989-90, Undated Copy, 4 pages.

D. DRAWINGS, PICTURES, ETC.

1. Copies of photographs (3), of Emergency Sump area.
2. Enclosure 4.11 from OP/1/A/1103/11, Elevations.
3. OC-CP-015-2 from OP/1,2,3/1103/11, Comparison of level plateaus for draining RCS.
4. Figure 1.1-9 (Partial), Reactor Bldg., General Arrangement, Sections.
5. Figure 5.1-9, Reactor Coolant System, Arrangement, Elevations.
6. OC-CP-015-1 from FSAR Fig 5.1.7, RCS Arrangement & Elevation.
7. OFD-102A-3.1, Revision 12, Flow Diagram of Low Pressure Injection System.
8. O-2435B, Revision 26, Unit 3 Piping Layout, Plan-Elevation 758'-0", Auxiliary Bldg.
9. OSFD102A-1, Revision 3, Summary Flow Diagram of Low Pressure Injection and Core Flood Systems.
10. Dwg. No. not shown, Unit 1 Piping Layout, Plant-Elevation, Auxiliary Bldg.

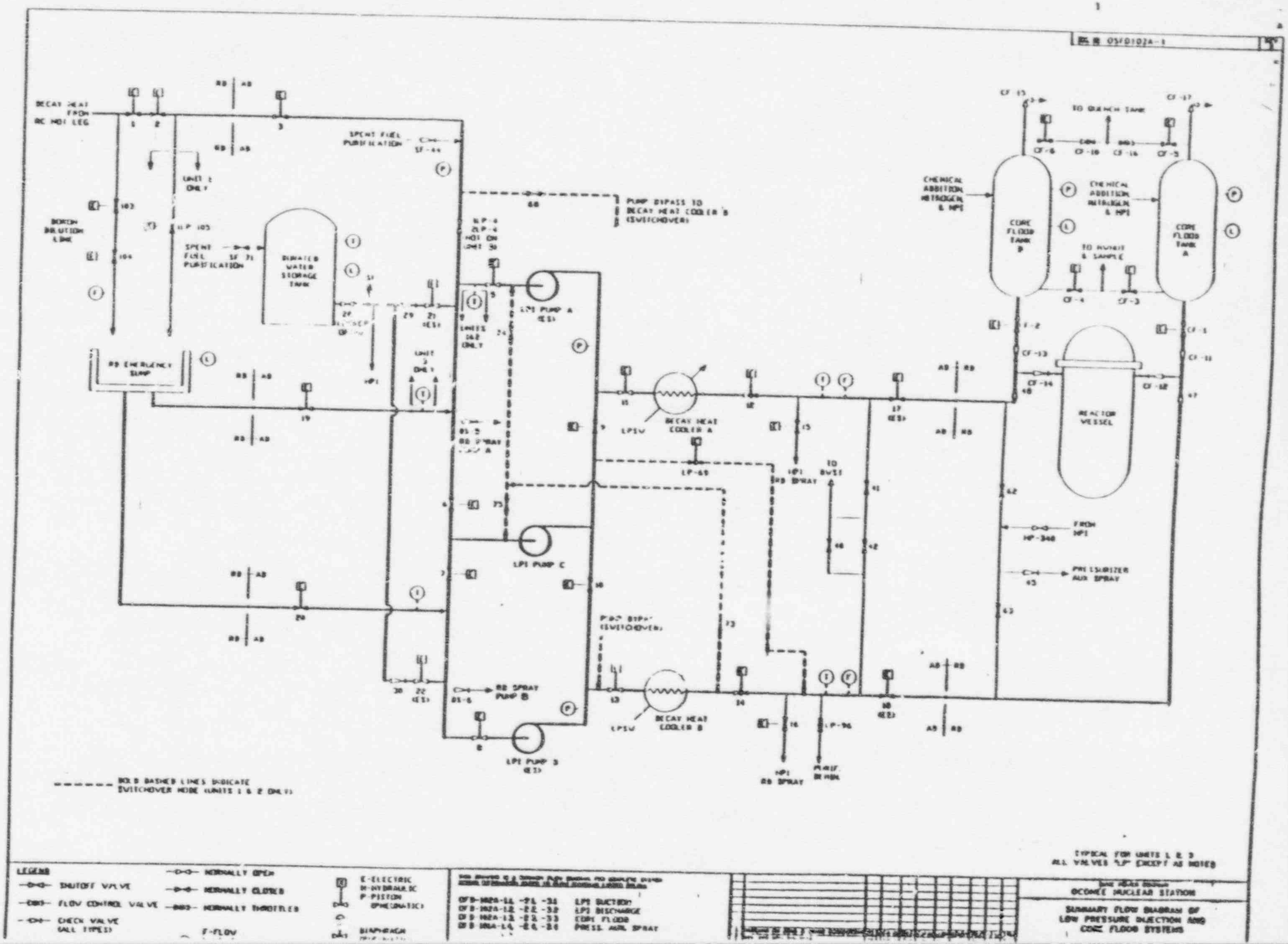
E. MISCELLANEOUS DOCUMENTS

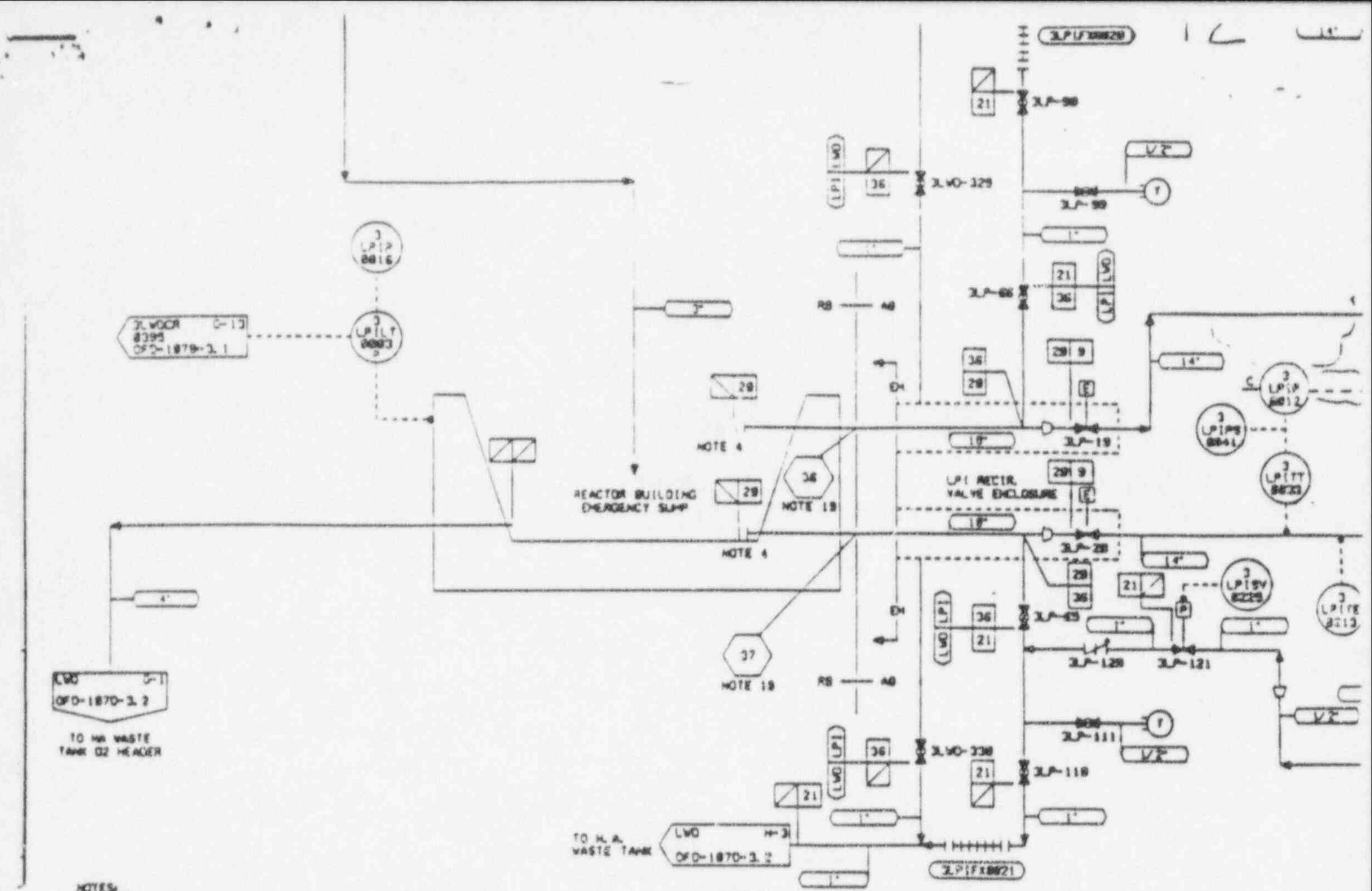
1. 10CFR50 Notification to the NRC (2 pages), for 3/8/91 event.
2. Loss of Decay Heat Removal Capability While Shutdown (93702), 3 pages, Sequence of Events on 3/8/91.
3. Sequence of Events for the Oconee Unit 3 Loss of Decay Heat Removal of March 8, 1991, 3 pages plus attachment.

4. Shift Incident Report, 4 pages.
5. Unit 3 SRO Log for 3/7, 3/8 and 3/9/91, 2 pages.
6. Shift Managers Log, pages 59 and 60.
7. RO Log for 3/8/91 from 0848 hrs to 0911 hrs, 1 page.
8. Oconee Nuclear Station Organization Charts, dated 12/31/90, 41 pages.
9. Operating Experience Item Review pertaining to IN 90-25 and NUREG 1410, 1 page.
10. Letter from L. Reed (Nuclear Operations) to R. L. White (Technical System Manager), dated 2/21/91, 5 pages.
11. File Memorandum prepared by L. Reed (Nuclear Operations), dated 10/4/90, 2 pages.
12. Letter from L. Reed (Nuclear Operations) to S. T. Rose (Technical System Manager I), dated 8/29/88, 15 pages.
13. Letter from G. B. Swindlehurst (Engineering Supervisor) to R. L. Sweigart (Superintendent of Operations), dated 1/11/89, 9 pages.
14. Intrastation Letter from G. K. McAninch (Integrated Scheduling), dated 11/21/88, 7 pages.
15. Letter from J. G. Torre (Regulatory Compliance) to H. B. Barron (Station Manager), dated 5/22/90, 13 pages.
16. Outage Manager Schedule for Unit 3, EOC-12 Refueling Outage, dated 3/9/91

FIGURE 1

LPI SYSTEM





NOTES:

1. OPERATING MODE REPRESENTED BY BOLD LINES: NORMAL DECAY HEAT REMOVAL.
2. VENT LINE TO BE TAKEN OFF AT HIGHEST POINT OF 12" SUCTION LINE.
3. ALL PIPING FROM BWST TO THE LP & BS PUMP SUCTIONS SHALL COMPLETELY DRAIN TO THE PUMP SUCTIONS BY GRAVITY.
4. BLIND FLANGE MUST BE REMOVED BEFORE START-UP.
5. 3LP-91 HAS REMOTE MANUAL OPERATOR FOR POST ACCIDENT ACCESSABILITY. SEE O-2437A.
6. TRACE HEAT OUTSIDE YARD PORTION OF PIPE AND SIX FEET INSIDE BUILDING WALL.
7. 1/2" THRU 6" SCH. #160
   8" THRU 18" SCH. #140
8. 1/2" THRU 2" SCH. #400
   2 1/2" THRU 12" SCH. #105
   14" THRU 24" SCH. #10
9. 1/2" THRU 8" SCH. #160
   12" SCH. #140
10. 1/2" THRU 3" SCH. #400
    4" THRU 12" SCH. #105
    14" THRU 18" SCH. #10
    19" SCH. #20
11. 18" SCH. #20
12. 1/2" THRU 3" SCH. #400
    4" THRU 6" SCH. #100
    8" THRU 12" SCH. #200
    14" SCH. #10
    18" THRU 18" SCH. #20
    1/2" THRU 8" SCH. #400

CONT'D

17. ORIFICE TO BE MADE BY DRILLING A 0.375" HOLE IN 12" PIPE AT THE HALF COUPLING CONNECTION.
18. VALVE DESIGN PRESSURE AND TEMPERATURE WILL BE PER UPSTREAM LINE NO. CLASS PER DOWNSTREAM LINE NO.
19. FOR DETAILS OF PENETRATION SCHEDULE TRANSITION REFER TO O-2438C
20. ORIGINAL ISSUE OF THIS DWG. WAS BASED ON PO-102A-3, REV. 30

LINE NO	DESIGN PRESS (PSIG)
1	2500
6	470 & 505
9	300
10	300
11	300
12	200
15	150
17	100
20	BLDG
21	BLDG
34	300
36	BLDG
42	2500
43	ATM
44	2500
46	470 & 505
48	2050
50	300
51	200
52	470 & 505
55	ATM
56	ATM
[illegible]	ATM

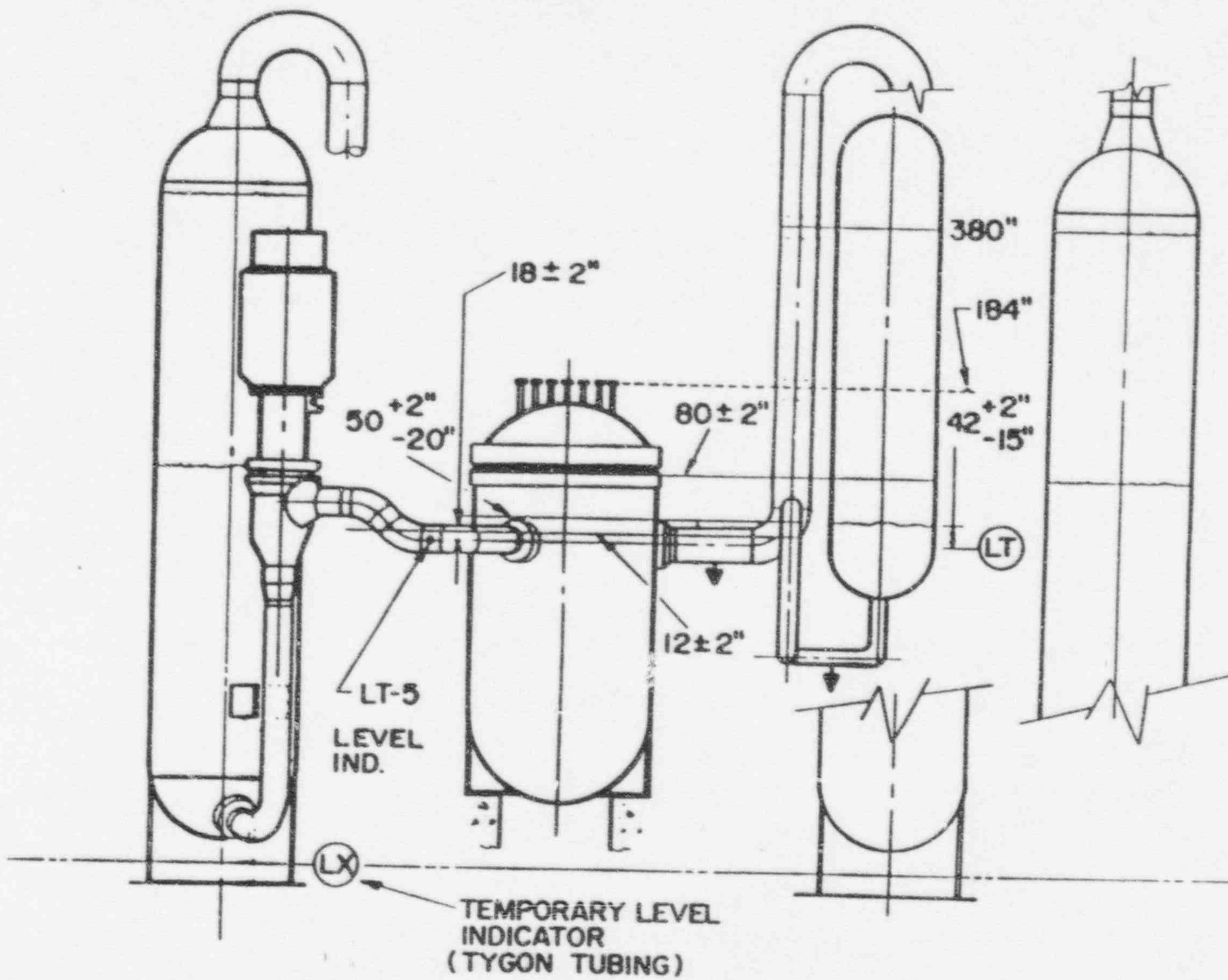
FIGURE 2

OFD-102A-3.1

Flow Diagram for Low-Pressure Injection System (Partial)

FI(3

LT-5 LOCATION



TITLE: RCS FILL, VENT, AND DRAIN	NOTES: COMPARISON OF LEVEL PLÂTEAUS FOR DRAINING RCS	ID NO OC-CP-015-2	DATE: 4/13/88
		REF OP/1,2,3/1103/11	
		DRN BY: WHC / RWP	APR BY RPG
		TRAINING USE ONLY	

FIGURE 4

STATUS OF GENERIC LETTER 88-17 COMMITMENT ITEMS DURING THE LOSS OF DECAY HEAT REMOVAL EVENT OF 3/8/91

COMMITMENT	PLANT STATUS
1. Containment closure survey conducted and containment closure is achievable within 2.5 hours.	1. Containment closure survey was not conducted but the ability to achieve containment closure within 2.5 hours existed.
2. Two operable core exit thermocouple indications and alarms are present.	2. No core exit thermocouples were operable.
3. LT-5 reactor vessel level indication system is available and operable.	3. LT-5 reactor vessel level indication system was available and operable.
4. An ultrasonic reactor vessel level detection system is available.	4. Ultrasonic reactor vessel level detection system was available.
5. Two Low Pressure Injection (LPI)pumps are operable to one or more operable LPI headers.	5. Two Low Pressure Injection pumps were operable to two headers.
6. Both Main Feeder Buses are energized.	6. Both Main Feeder Buses were energized.
7. Two sources of power shall be available to supply the Main Feeder buses.	7. Startup transformer CT-3 and Standby transformers CT-4 and CT-5 were available to supply MFBs.
8. Two of the following means of adding inventory to the RCS is available: a) BWST gravity flow b) One Bleed Transfer Pump c) One High Pressure Injection (HPI) Pump	8. The status of flow paths for adding RCS inventory was as follows: a) BWST gravity flow available. b) No Bleed Transfer Pumps available. (piping disassembled) c) No HPI pumps available.
9. Both steam generator upper primary side handhole covers or equivalent RCS vent path shall be removed.	9. Reactor Vessel head and pressurizer relief valve were removed.
10.Testing and maintenance activities are reviewed for adverse effects on decay heat removal systems.	10. Testing and maintenance activities were not specifically being reviewed for adverse decay heat removal effects.

FIGURE

OCONEE UNIT 3 LOSS OF DECA EAT REMOVAL EVENTS AND CAUSAL FAC RS CHART