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

Report Nos.: 50-413/92-18 and 50-414/92-18

Licensee: Duke Power Company
P.O. Box 1007
Charlotte, N.C. 28201-1007

Docket Nos.: 50-413 and 50-414 License Nos.: NPF-35 and NPF-52

Facility Name: Catawba Nuclear Station Units 1 and 2

Inspection Conducted: July 12, 1992 - August 8, 1992

Inspector: W. H. Miller Jr. 9-3-92
For W. T. Orders, Senior Resident Inspector Date Signed

Inspector: W. H. Miller Jr. 9-3-92
For P. C. Hopkins, Resident Inspector Date Signed

Inspector: W. H. Miller Jr. 9-3-92
For J. Zeiler, Resident Inspector Date Signed

Approved by: George A. Belisle 9-3-92
George A. Belisle, Chief Date Signed
Projects Section 3A
Division of Reactor Projects

SUMMARY

Scope: This routine, resident inspection was conducted in the areas of plant operations review; security badge access code errors; review of decay heat removal/shutdown risk; inadequate corrective actions concerning operator response to BDMS actuations; inadvertent chemical release; surveillance observations; maintenance observations; follow-up of previous inspection findings; and information meeting with local off[illegible]ls.

Results: One violation for failure to follow procedures was identified and involved inadequate operator response to a Boron Dilution Mitigation System (BDMS) actuation (paragraph 6). This violation is similar to a violation that was identified on June 4, 1991.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

*S. Bradshaw, Shift Operations Manager
J. Forbes, Engineering Manager
S. Frye, Operations Support Manager
*R. Futrell, Regulatory Compliance Manager
E. Geddie, Operations Superintendent
T. Harrall, Safety Assurance Manager
M. Haz tine, Compliance
*J. Lowery, Compliance
W. McCollum, Station Manager
*R. Propst, Chemistry Manager
M. Tuckman, Catawba Site Vice-President

Other licensee employees contacted included technicians, operators, mechanics, security force members, and office personnel.

NRC Resident Inspectors

*W. Orders
P. Hopkins
*J. Zeiler

* Attended exit interview.

2. Plant Status

Unit 1 Summary

Unit 1 began the report period in Mode 4, in day 1 of a planned 71 day End-of-Cycle 6 refueling outage. On July 12, the unit entered Mode 5, Cold Shutdown. Major activities initiated or accomplished while in Mode 5 included reactor coolant (NC) system crud burst and cleanup, ice condenser refrigeration modifications, and NC system drain to Midloop conditions. On July 20, the unit entered Mode 6, Refueling. Defueling of the reactor core commenced on July 25 and was completed on July 29 without incident. Major outage activities initiated or accomplished while defueled included; steam generator eddy current testing, Diesel Generator 1A inspection and maintenance, ice condenser maintenance, valve testing maintenance, and main turbine/generator maintenance.

Unit 2 Summary

Unit 2 began the report period at full power. On July 13, power was reduced to 65 percent following the identification of a hydraulic leak on a main turbine control test valve. The leak was repaired and the unit returned to full power the following day. The unit operated at essentially full power the remainder of the report period.

3. Plant Operations Review (71707)

The inspectors reviewed plant operations throughout the report period to verify conformance with regulatory requirements, Technical Specifications (TS) and administrative controls. Control Room logs, the Technical Specification Action Item Log, and the Removal and Restoration (R&R) log were routinely reviewed. Shift turnovers were observed to verify that they were conducted in accordance with approved procedures. The complement of licensed personnel on each shift inspected, met or exceeded the requirements of Technical Specifications. Further, daily plant status meetings were routinely attended.

Plant tours were performed on a routine basis. The areas toured included but were not limited to the following:

Turbine Buildings
Auxiliary Building
Units 1 and 2 Diesel Generator Rooms
Units 1 and 2 Vital Switchgear Rooms
Units 1 and 2 Vital Battery Rooms
Standby Shutdown Facility

During the plant tours, the inspectors verified by observation and interviews that measures taken to assure physical protection of the facility met current requirements. Areas inspected included the security organization, the establishment and maintenance of gates, doors, and isolation zones in the proper conditions, and that access control badging was proper and procedures were being followed.

In addition, the areas toured were observed for fire prevention and protection activities and radiological control practices. The inspectors also reviewed Problem Investigation Reports (PIRs) to determine if the licensee was appropriately documenting problems and implementing corrective actions.

No violations or deviations were identified.

4. Security Badge Access Code Errors (71707)

On July 15, 1992, during a routine periodic review of the Catawba security badge summary, the resident inspector detected that 2 NRC Region II inspectors had apparently been on site on July 8. The inspector knew that those inspectors had not been on site and asked the licensee's security group to investigate the reason for the erroneous information.

The licensee's investigation revealed that due to a known problem with the security badges used at Catawba, the badges may read other personnel's identification/access codes. One possible result of this inadequacy could be personnel being able to access areas for which they are not authorized.

The inspector also determined that the licensee did not have a program in place for periodic card verification. The licensee subsequently stated that they realize the need for such a program and were working expeditiously to implement one.

It should be noted that this problem is generic to all three Duke nuclear facilities.

This item is identified as Inspector Followup Item (IFI) 413, 414/92-18-01, Security Badge Access Code Errors, and will be evaluated by Region II Security personnel.

No violations or deviations were identified.

5. Review of Decay Heat Removal/Shutdown Risk (TI 2515/[illegible])

The resident inspectors began a routine rev[illegible] [illegible]es implemented by the licensee to reduce the risks associ[illegible] [illegible]ations conducted during reduced reactor coolant (NC) invent[illegible]

The inspectors reviewed the training package developed for the l[illegible] staff on associated [illegible] mid-loop events. Information brief[illegible] concerning implications of these events were also conducted for appropriate non-licensed personnel. The inspectors audited these presentations as well as the training which was completed before the unit entered a condition of reduced inventory

The inspectors also reviewed controls implemented to establish containment closure. These controls include a pre-established barrier similar to containment integrity. Containment closure is established prior to entry into a condition of reduced NC System inventory. Procedures establish the boundary and require a continuous status of each penetration. This is controlled by a dedicated reactor operator (RO). The RO's responsibilities also include reviewing R&Rs that may affect containment penetration status, updating the Penetration Status Board in the Control Room and monitoring plant indications to insure that they accurately reflect containment penetration status.

Reactor coolant system core exit temperature indications are provided by four thermocouples which remain attached until just prior to head removal. These temperatures are indicated on the operator aid computer (OAC) which automatically and continuously monitors core exit temperatures. Alarms are set to insure that developing trends are detected. The temperatures are also plotted on a chart recorder to provide recorded trending information. The use of the OAC with temporary alarms and hard wired alarms on Residual Heat Removal (ND) pump discharge temperature instrumentation, NC level instrumentation, source range controls, and pressurizer levels adds a higher degree of confidence and accuracy to mid-loop activities.

With respect to assured sources of NC makeup, several means of inventory makeup have been identified. The Unit Outage Coordinators protect at least two of these flowpaths during all phases of the outage and identify the available flowpaths to the Control Room operators. This information is reviewed at each shift turnover to insure the latest system status is maintained. Adequate procedures are in place to provide the operators information on the use of makeup flowpaths.

During reduced inventory activities, tailgate meetings are conducted with each shift and other involved personnel. The meetings provide a review of industry mid-loop incidents and an overview of total mid-loop and core operations, such as precautions, impact on TSs, procedure summaries, expected alarms and annunciators.

The residents will continue this review and will document the results in NRC Inspection Report Nos. 50-413, 414/92-22.

No violations or deviations were identified.

6. Inadequate Operator Response To BDMS Actuations (71707)

On July 17, 1992, Catawba Unit 1 was in Mode 5 in day 6 of a planned 71 day refueling outage. At 1:53 p.m., the Unit experienced an actuation of train A of the Boron Dilution Mitigation System. This system is designed to mitigate an inadvertent dilution of the Reactor Coolant System (NC). One automatic function of the system is to realign the centrifugal charging pumps (NV) from the volume control tank (VCT) to the refueling water storage tank (FWST) in order to inject highly borated (2000 ppm) water into the NC system. The BDMS system accomplishes this realignment by closing the normal flowpath valve NV-188 from the VCT and opening valve NV-252 from the FWST.

In responding to the BDMS actuation alarm, the control room operator believing the actuation to be spurious, closed valve NV-252, but failed to open valve NV-188. After having performed this action, he referred to the alarm response procedure to determine if the action he had taken was correct. It was at this point that the operator realized that valve NV-188 should have been opened. As a result of the operator's error, the 1B charging pump ran for approximately 40 seconds with no suction source.

Operations Management Procedure (OMP) 1-8, Authority and Responsibility of Licensed Operators and Licensed Senior Reactor Operators, Section 7.2.B, describes the responsibilities of the OATC. Step 7.2.B.9.c of the procedure requires that the OATC verify that the appropriate automatic actions for the alarm have taken place prior to taking recovery action. The OATC failed to refer to the annunciator response procedure, OP/1/B/6100/10C, Annunciator Response For Panel 1AD-2, which details the automatic actions for the BDMS alarm until after he had taken action to realign the system.

Technical Specification 6.8.1 requires in part that written procedures be established, implemented and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February

1978. OMP 1-8, Step and 7.2.B.9.c delineates the actions to be performed by control room personnel regarding responses to annunciators. The OATC did not reopen the suction valve from the VCT and did not refer to the annunciator response procedure to verify that the automatic actions for the BDMS alarm had taken place until after he had taken action to recover from the actuation. This failure to follow procedure OMP 1-8 is identified as Violation 413/92-18-02: Inappropriate Operator Response to BDMS Alarm Actuation.

It should also be noted that this event is virtually identical to one which occurred on June 4, 1991. This was during the last Unit 1 refueling outage when Train A of the BDMS actuated causing the suction for the operating 1A NV Pump to realign from the VCT to the FWST. In that event, the operator closed the suction valve from the FWST but failed to reopen the suction valve from the VCT. The pump ran for approximately 17 minutes without a source of water before the operator realized that the suction valve from the VCT was closed.

In both events, the operators failed to take required actions in accordance with procedures and also failed to verify that appropriate automatic actions had taken place as required by Operations Management Procedure (OMP) 1-8.

The event of June 4, 1991, was cited in NRC Inspection Report Nos. 50-413,414/91-15 as a violation of Technical Specification (TS) 6.8.1 in that the operator had failed to refer to the BDMS alarm response procedure as described above.

In a letter dated August 29, 1991, in response to the Notice of Violation, the licensee detailed the corrective actions which had been or would be taken to preclude the recurrence of the event.

The proposed corrective actions were:

a) "The correct procedure on responding to an unexpected alarm has been re-emphasized to the Control Room Operator involved in this incident. The correct procedure for responding to an unexpected alarm has been reinforced to all licensed operators through the issuance of an operator update on this incident."

b) "This incident will be covered during operator requalification training including the automatic actions which occur on a BDMS alarm. Modifications to the BDMS will be performed to eliminate excessive spurious alarms by April 1, 1992."

This inspectors verified that the corrective actions had been completed and noted that the corrective actions appeared to be ineffective. This issue was discussed with the licensee during the exit interview because recurring violations are of particular concern to the NRC. It is expected that licensees learn from their past failures take effective corrective action to preclude recurrence.

One violation was identified.

7. Inadvertent Chemical Release (71707)

On July 22, 1992, at approximately 10:30 a.m., a conventional waste water treatment system pipe from the turbine building sump (WP) discharge line was over-pressurized causing the pipe to rupture. The result was the discharge of approximately 1.33 pounds of hydrazine to the environment.

At the time of the event, Unit 2 was at 100 percent power and Unit 1 was in Mode 6 for the EOC6 refueling outage. As part of the outage schedule, water containing hydrazine had been drained from the Unit 1 hotwell to the Unit 1 Turbine Building Sump (WF). Maintenance had been scheduled on the Service Building Sump (WB) System to replace valve 1WB30, and to open the spectacle flange which had been installed per a Temporary Station Modification (TSM). Procedure OP/0/B/6500/12, Enclosure 4.9, Rotating the Spectacle Flange to Isolate WB from WP Header, specifies actions that were to have been sequentially performed for that evolution. These actions included the following:

1.1 Have R&Rs written by Operations to tag the Unit 1 and 2 turbine building sumps in the OFF position.

1.2 Have R&Rs written by Chemistry to tag 1WB-30 Closed.

1.3 Have R&Rs written by Chemistry to tag 1WC-162 Closed.

1.4 Have R&Rs written by Chemistry to open 1WC-144.

1.5 Have Maintenance pre-stage tools and equipment in the Water Treatment Room.

1.6 Upon completion of Steps 1.1 through 1.6, notify Operations to pump down the Unit 1 and 2 sumps to the lowest level possible and hang their red tags.

1.7 Have Chemistry hang their red tags.

Procedure OP/0/B/6500/12 is not a safety-related procedure. The work activity was not being performed on safety-related systems. However, this should not negate overall management directives for performing work correctly. In this work activity, not only were these actions not performed in sequence, an R&R was not written for step 1.4. The ultimate result was that valves 1WB30, 1WB29 and 1WC162 were closed isolating the service building sump system, but the turbine building sump pumps had not been de-energized. Consequently, when a turbine building sump pump started, the system was over-pressurized and the discharge lines ruptured. This resulted in an inadvertent chemical release to the environment. The licensee notified the appropriate agencies because of the inadvertent chemical release.

8. Surveillance Observations (61726)

a. General

During the inspection period, the inspectors verified plant operations were in compliance with various TS requirements. Typical of these requirements were confirmation of compliance with the TS for reactivity control systems, reactor coolant systems, safety injection systems, emergency safeguards systems, emergency power systems, containment, and other important plant support systems. The inspectors verified that: surveillance testing was performed in accordance with approved written procedures, test instrumentation was calibrated, limiting conditions for operation were met, appropriate removal and restoration of the affected equipment was accomplished, test results met acceptance criteria and were reviewed by personnel other than the individual directing the test, and any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

b. Surveillance Activities Reviewed

The inspectors witnessed or reviewed the following surveillances:

PT/1/A/4200/02C-H	Containment Closure Verification
PT/1/A/4200/41C	Containment Purge Automatic Valve
PT/1/A/4550/01D	Reactor Building Manipulator Crane load test
PT/0/A/4150/24	Fuel Assembly Examination
PT/0/A/4150/18	Fuel Assembly Insert Shuffle
PT/0/A/4550/09	Fuel Assembly Insert Shuffle for next cycle
PT/0/A/4150/17	Fuel Assembly Core Unload For Next Cycle
PT/2/A/4200/01L	Controlling Procedure for Type B & C Leak Rate Test
PT/2/A/4200/09A	Auxiliary Safeguards Test Cabinet Periodic Test
PT/2/A/4200/41A	Containment Purge Isolation Valve Leak Rate Test
PT/2/A/4450/05B	Containment Air Return Fan and Hydrogen Skimmer Fan 2B Performance Test

No violations or deviations were identified.

9. Maintenance Observations (62703)

a. General

Station maintenance activities of selected systems and components were observed/reviewed to ensure that they were conducted in accordance with the applicable requirements. The inspectors verified licensee conformance to the requirements in the following areas of inspection: activities were accomplished using approved procedures, and functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities performed were accomplished by qualified personnel; and materials used were properly certified. Work requests were reviewed to determine the status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which may affect system performance.

b. Maintenance Activities Reviewed

The inspectors witnessed or reviewed the maintenance activities covered by the following Work Requests (WRs):

WR 92018289	Remove and repack Valve 1HW40
WR 92012068	Remove and repack Valve 1HW-2 heater drain
WR 92020298	Vital Battery charger (essential charger) capacity test
WR 92047057	Inspection of CBB Battery Connections
WR 92021576	2EPK inspections and clean cells
WR 92009246	Perform electrical/electronics inspections work on turbine generator
WR 92001117	Inspect "A" Feedwater pump bearing
WR 92002288	Eddy current testing and inspection of Feedwater heaters
WR 92017654	Teardown and inspect turbine generator
WR 92010062	Verify mechanical integrity of Diesel Generator Cooling Water Valve 1KD-6
WR 91096299	Replace Diesel Generator Jacket Water Pump Drive Gear
WR 92015178	Perform 10-year overhaul inspection of Diesel Generator 1A

No violations or deviations were identified.

10. Followup on Previous Inspection Findings (92701 and 92702)

(Open) Unresolved Item (URI) 413, 414/92-17-02: Review Licensee's Non-Conservative Change to Control Room Dose Calculation.

In a previous inspection, a concern was raised involving the licensee's removal of the ECCS leakage source term from the control room operator dose analysis in 1989. Removal of this term reduces the total control room operator dose assumed for a design-basis LOCA. This lower value of control room operator dose was subsequently used in compensatory actions implemented when the control room pressure boundary was degraded. During this report period, the licensee completed a past operability review of these compensatory actions incorporating ECCS leakage in the dose analysis. Based on this review, it was determined that for the compensatory actions, the control room operator dose limit of 30 Rem, as defined in General Design Criterion (GDC) 19, would not have been exceeded.

The basis for including ECCS leakage in the control room operator dose analysis for a design-basis LOCA is contained in NUREG-0737, Clarification of TMI Action Plan, Item III.D.3.4.3, Control Room Habitability Requirements. Item III.D.3.4.3 it states that the radiation source term should be for the LOCA containment leakage and ESF leakage contribution outside containment (i.e., ECCS leakage) as described in Appendix A and B of Standard Review Plan (SRP) Chapter 15.6.5.

The licensee stated that in 1989 the ECCS leakage term was removed following a re-analysis of the control room dose methodology to support a proposed control room compensatory action for planned maintenance that degraded the control room ventilation boundary area. This review was performed under the Design Engineering QA Program as required by 10 CFR 50, Appendix B, Criterion III. The reviewer used NUREG-0800, Standard Review Plan (July 1981), Section 6.4, Control Room Habitability, to perform the re-analysis. Since Section 6.4 did not specifically discuss the inclusion of ECCS leakage in the control room operator dose, and since it was stated that if Section 6.4 was followed, NUREG-0737, Item III.D.3.4.3 would be satisfied, the reviewer did not specifically review Item III.D.3.4.3. The licensee stated that the guidance provided in NUREG-0800 was not sufficiently clear to ensure that ECCS leakage would be included in the control room operator dose as intended by NUREG-0737 and questioned whether NUREG-0800 intended the inclusion of ECCS leakage.

The inspectors reviewed the Catawba FSAR (1989 Update) to determine what assumptions were used in the control room operator dose analysis. In Section 15.6.5.3, Environmental Consequences, which describes the radiological dose consequences of a LOCA for offsite personnel and control room operators, it was stated that certain parameters and assumptions used in the offsite dose analysis were also used for the control room operator dose analysis. One of these offsite dose assumptions was that ECCS leakage occurs at twice the maximum operational leakage. Therefore, it appeared that the FSAR confirmed that ECCS leakage was considered as an assumption in the control room operator dose analysis. If this is the case, the inspectors believed that the inclusion of ECCS leakage was part of the licensing basis for the facility. In order to delete this term, the licensee should have performed a 10 CFR 50.59 review since this would represent a change to the facility as described in the FSAR. The licensee did not agree that removing the ECCS leakage term constituted a change to the facility as described in the FSAR.

The inspectors also reviewed the licensee's outside containment leakage reduction program as it relates to ensuring that the ECCS leakage assumptions used in the offsite and control room operator dose analyses are not exceeded. Technical Specification 6.8.4.a requires a leakage program be established to include: 1) preventive maintenance and periodic visual inspection, and, 2) integrated leak test requirements of appropriate systems at refueling cycle intervals or less. The licensee performs a weekly visual inspection of systems carrying radioactive fluids outside containment. Work Requests are initiated for identified leaks. In addition, every 18 months these systems are operated and leakage checks are performed. Although Work Requests are initiated for identified leaks, depending on the severity and accessibility of the leak, these items are not always expeditiously corrected. Since no attempt is made to quantify the leakage, the inspectors questioned how the licensee ensured that the ECCS leakage value that was assumed in the offsite and control room operator dose analyses was not exceeded. The licensee questioned whether the intent of TS 6.8.4 was to quantify this leakage and compare it against the dose analysis assumptions.

Based on the controversy over whether ECCS leakage is required to be included in the control room operator dose analysis, and the adequacy of the licensee's implementation of an outside containment leakage reduction program, the inspectors determined that these issues should be referred to NRC management for review and resolution.

The results of the NRC management review and resolution of these issues will be reported in a future inspection report. This URI remains open pending the resolution of these issues.

No violations or deviations were identified.

11. Information Meeting with Local Officials (94600)

A meeting was held with North Carolina, South Carolina, and Gaston County local officials on July 14, 1992. This meeting was conducted to familiarize the local officials with mission of the NRC, to introduce key NRC personnel associated with NRC Region II and Catawba, to discuss lines of communications available between local officials and the NRC, and to discuss the status of the facility and related community concerns with these local officials. A copy of the general information was provided to the attendees.

12. Exit Interview

The inspection scope and findings were summarized on August 12, 1992, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection findings listed below. No dissenting comments were received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

Item Number	Description and Reference
IFI 413, 414/92-18-01	Security Badge Access Code Errors (paragraph 4).

VIO 413/92-18-02	Failure to Follow Procedures, Inadequate Operator Response to BDMS Actuations (paragraph 6).