UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

January 22, 2003

Garry L. Randolph, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

SUBJECT: NRC INSPECTION REPORT 50-483/02-06

Dear Mr. Randolph:

On December 28, 2002, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on January 6, 2003, with Mr. Ron Affolter, Vice President, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified an issue that was evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that a violation is associated with this issue. This violation is being treated as a noncited violation (NCV), consistent with Section VI.A of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Callaway Plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

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NRC Inspection Report
50-483/02-06

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report:	50-483/02-06
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	October 6 through December 28, 2002
Inspectors:	M. S. Peck, Senior Resident Inspector J. D. Hanna, Resident Inspector D. R. Carter, Health Physicist L. E. Ellershaw, Senior Reactor Inspector C. J. Paulk, Senior Project Engineer F. L. Brush, Senior Resident Inspector
Approved By:	D. N. Graves, Chief, Project Branch B

SUMMARY OF FINDINGS

Callaway Plant
NRC Inspection Report 50-483/02-06

IR 05000483-02-06; Union Electric Co; on 10/06-12/28/2002; Callaway Plant. Integrated Resident & Regional Report; Personnel Performance During Nonroutine Evolutions and Events.

The report covers a 12-week period of routine Resident and Regional inspection activities from October 6 through December 28, 2002. One finding of significance was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Cornerstone: Initiating Events

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criteria III, Design Control, occurred when the licensee failed to maintain control of the over temperature-delta temperature delta flux penalty circuit amplifier gain.

  The finding was greater than minor because the condition resulted in a transient initiator and contributed to an unplanned reactor trip, an initiating event. This finding was evaluated using Appendix A of the reactor safety significance determination process and determined to be of very low safety significance because the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident, did not contribute to both the likelihood of a reactor trip and the unavailability of mitigation equipment, and did not increase the likelihood of a fire or flood. This finding is in the licensee's corrective action system as Callaway Action Request System Number 200208352 (Section 1R14.2).

Report Details

Summary of Plant Status: At the beginning of the inspection period, Union Electric Company was operating the facility at full power. The licensee reduced power to 80 percent on October 21, 2002, to support on-line main steam safety valve testing followed by a reactor shutdown on October 23 for Refueling Outage 12. The licensee completed refueling activities and restarted the reactor on November 26. The licensee operated the facility at full power until an unplanned automatic reactor shutdown on December 14. The licensee restarted the reactor on December 17 and operated the plant at full power until the end of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's adverse weather preparations to verify that design features and procedural implementation adequately protected mitigating systems. The inspectors walked down the borated refueling water storage system during October 2002, to verify that cold weather would not degrade the system function or operability. The inspectors also reviewed licensee Off-Normal Procedure OTN-BN-00001, "Borated Refueling Water Storage System," Revision 5, and discussed adverse weather preparations with the licensee.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors completed three partial system walkdowns of safety significant equipment during the inspection period. The inspectors performed the walkdowns to verify proper component alignment and equipment readiness when redundant systems were removed from service for maintenance or testing. The inspectors completed the following partial walkdowns during the quarter:

- Train A Residual Heat Removal: The inspectors completed a walkdown of Train A of the residual heat removal subsystem to verify correct mechanical and electrical alignment of components located in the auxiliary and control buildings. The inspectors completed the walkdown on November 11, 2002, while the redundant residual heat removal subsystem was out of service for maintenance. The inspectors reviewed critical portions of system alignments using the Final Safety Analysis Report, Section 6.2.2, "Containment Heat Removal Systems," and Drawing M-22EJ01, "Piping and Instrumentation Diagram, Residual Heat Removal System," Revision 47.

- Alternate Off-Site Power Supply: The inspectors walked down power distribution to plant safety-related loads while the off-site power was backfed through the main transformers. The inspectors completed the walkdown of turbine building, auxiliary building, and control building components on November 8, 2002, while the startup transformer was out of service for maintenance. During the walkdown, the inspectors compared the plant electrical alignment with Worker Protection Assurance System Numbers 42659, 42854, and 42853; Electrical Drawing E-21001, “Main Single Line Diagram,” Revision 10; Technical Specification 3.8.2, “AC Sources - Shutdown”; and Final Safety Analysis Report, Section 8.3.1, “AC Power Systems.”

- Condenser Steam Dump System: The inspectors walked down a portion of the condenser steam dump system in the turbine building and the control room to verify correct mechanical and electrical alignment. The inspectors completed the walkdown on December 9, 2002, while low pressure condenser main steam dump control Valves ABUV-0034, -0035, and -0036 were out of service for maintenance. The inspectors reviewed critical portions of system alignments using Drawing M-22AB03, “Piping and Instrumentation Diagram Main Steam System,” Revision 12.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown of the Containment Spray System

a. Inspection Scope

The inspectors performed a walkdown of the containment spray system on December 20 and 23, 2002, to verify correct mechanical and electrical component alignment. During the walkdown, the inspectors also reviewed component leakage that may impact system function, labeling, lubrication, and cooling and verified that hangers and supports were correctly installed and functional and that essential support systems were operational. The inspectors used Drawing M-22EN01, “Containment Spray System,” Revision 6; Drawing E-21NB01, “Lower Medium Voltage System Class IE Single Line Meter and Relay Diagram,” Revision 7; Finial Safety Analysis Report, Section 6.2.2, “Containment Heat Removal Systems”; Technical Specification 3.6.6, “Containment Spray and Cooling systems”; and Technical Specification 3.6.7, “Recirculation Fluid pH Control System,” to determine system requirements.

The inspectors also assessed operability and conformance of the containment spray system with licensing requirements and commitments by in-office review. The inspectors considered the licensee’s corrective measures to address Callaway Action Request System Number 200207243, “Containment Sump Inspection Action Items,” November 6, 2002. The inspectors used the Final Safety Analysis Report, Section 6.2.2, “Containment Heat Removal Systems”; Technical Specification 3.6.6, “Containment Spray and Cooling Systems”; Technical Specification 3.6.7, “Recirculation

Fluid pH Control System"; and System Operating Procedure 2.4.6, "Reactor Core Isolation Cooling System," Revision 33, as the bases for acceptance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed six fire protection walkdowns to verify operational status and material condition of fire detection and mitigation systems, passive fire barriers, and suppression equipment. The inspectors also reviewed the licensee's implementation of combustible material controls and ignition sources in selected fire protection zones. The inspectors compared plant conditions against descriptions and commitments described in the Final Safety Analysis Report, Section 9.5.1, "Fire Protection System," and Appendix 9.5B, "Fire Hazard Analysis." The inspected areas included:

- Fire Area F-1A, fuel building, 2000 foot elevation, completed on November 1, 2002
- Fire Area RB-1, reactor building, 2001 foot elevation, completed on November 1, 2002
- Fire Area D-1, diesel generator building, 2000 foot elevation, completed on November 7, 2002
- Fire Area A-8, auxiliary building, 2000 foot elevation, general area, completed on November 11, 2002
- Fire Areas A-9 and A-10, residual heat removal heat exchanger rooms, completed on November 11, 2002
- Fire Areas A-13, A-14, and A-15, auxiliary feedwater pump and valve rooms, completed on November 18, 2002

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the causes and corrective actions associated with a past plant internal flooding event which led to the loss of the central alarm station (Callaway Action Request System Number 200203108, "Power Lost to Central Alarm Station," May 12, 2002). The inspectors completed a walkdown of the central alarm station on December 9, 2002. The inspectors conducted the walkdown to verify that the licensee had implemented adequate corrective actions to prevent recurrence of the flood. The inspectors observed that equipment below the flood-line, including electrical conduits, holes, and wall penetrations, were sealed. The inspectors walked down the common drain system and sumps and observed operable sump pumps, level alarms, and control circuits. The inspectors also inspected underground bunkers and manholes, subject to flooding, that contain multiple pieces of equipment which support the security systems.

The inspectors used Engineering Change Notice RHR-22246A, Revision 9; Drawing 8600-X-89705, "Interconnecting Diagram Cable Routing," Revision 10; and Conduit and Equipment Drawings 8600-X-89325, "Security Building," Revision 2; and 8600-X-89323, "Security Building," Revision 13, as the bases for acceptability of the plant configuration.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination Activities

a. Inspection Scope

The inspectors observed and reviewed the completed documentation for the ultrasonic examination of main feedwater Weld 2-AE-04-F20, a pipe-to-valve circumferential weld, and a liquid penetrant examination of Field Weld FW-5, a flange-to-instrument line for flow Orifice EJFO0003. The inspectors also reviewed the completed documentation of the following examinations of the residual heat removal system.

Component/Weld Identification	Examination Method
2-EJ-05-S011-C, pipe-to-reducer circumferential weld	Ultrasonic
2-EJ-02-FW88, elbow-to-pipe circumferential weld	Ultrasonic
2-EJ-02-F020, pipe-to-tee circumferential weld	Ultrasonic
2-EJ-02-S001-C, elbow-to-pipe circumferential weld	Ultrasonic
2-EJ-02-F050, elbow-to-pipe circumferential weld	Ultrasonic
2-EJ-02-FW102, elbow-to-pipe circumferential weld	Ultrasonic

Component/Weld Identification	Examination Method
2-EJ-02-F031, pipe-to-tee circumferential weld	Ultrasonic
2-EJ-02-S023-B, pipe-to-tee circumferential weld	Ultrasonic
2-EJ-02-S022-B, pipe-to-weld circumferential weld	Ultrasonic
2-EJ-03-S003-F, elbow-to-pipe circumferential weld	Ultrasonic
2-EJ-02-S023-D, pipe-to-tee circumferential weld	Ultrasonic

In addition to the ultrasonic examination documents, the inspectors reviewed the following radiographic examination data sheets for shop welds.

Work Package	Weld Identification
W220063	FW-3
W220063	FW-4
W203577	FW-3
W203577	FW-3A
W203577	FW-4

During the review of each examination, the inspectors verified that the correct nondestructive examination procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed the documentation to determine if the indications revealed by the examinations were compared against the previous outage examination reports and the American Society of Mechanical Engineers (ASME) Code specified acceptance standards. This review was also performed to determine if the indications were appropriately dispositioned.

The inspectors also reviewed the repair and replacement activities associated with a leak on the upstream test connection for residual heat removal system flow Orifice EJFO0003 to assure that the appropriate regulatory and ASME Code requirements were met.

b. Findings

No findings of significance were identified.

.2 Performance of Steam Generator Tube Examination Activities

a. Inspection Scope

The inspectors reviewed the licensee's policy document, UEND-Strategy-02, Revision 3, "Steam Generator Strategic Plan for Callaway," which provided an approach to steam generator maintenance and a long-term strategy for steam generator operability. Procedures providing instructions and guidance for acquisition and analysis of steam generator eddy current data, personnel qualifications, equipment, calibration standards, and installation of tube plugs were reviewed against ASME Code requirements and accepted industry practices.

Electric Power Research Institute (EPRI) guidance regarding in situ pressure testing and Nuclear Energy Institute (NEI) steam generator program guidelines were reviewed by the inspectors. These documents, generally endorsed by NRC, provide guidelines regarding condition monitoring, operational assessments, and in situ pressure testing. The inspectors reviewed the licensee's methodology and screening protocols for sample size and selection of tubes for in situ pressure testing to ensure the most limiting tubes, from a structural and accident-induced leakage integrity standpoint, are included. Further, the inspectors reviewed the Callaway Plant degradation and operational assessments performed by Framatome ANP (degradation - Refueling Outage 12) and Dominion Engineering, Inc. (operational - Cycle 12), and noted that the predicted tube degradation conclusions were similar to what was being identified during the eddy current examinations performed during this inspection.

The inspectors reviewed three of the Examination Techniques Specification Sheets (ETSS) that were qualified for use at the Callaway Plant and the applicable procedure qualification records and determined if they were in accordance with EPRI guidelines. Additionally, during the inspection the inspectors observed system calibration, eddy current data acquisition, and steam generator tube plugging activities.

During the inspection, the inspectors reviewed actions taken by the licensee as a result of Steam Generator A being placed into the C-3 category for failing to meet the acceptance criteria regarding defective tubes. This included Technical Specification required actions pertaining to increased sample size and test location and the resultant notification to the NRC (Event Notification 39345).

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

In addition to the examinations observed and reviewed, the inspector reviewed the two condition reports issued during the past year on inservice inspection activities. The reviews were performed to determine if the licensee employees identified, evaluated, corrected, and trended problems.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalifications (71111.11)

a. Inspection Scope

The inspectors observed three licensed operator training exercises to identify deficiencies and discrepancies and to assess operator performance and the evaluator's critique. The inspectors placed emphasis on high-risk licensed operator actions, activities associated with the emergency plan, previous lessons learned items, and plant experiences. The inspectors used Operating Procedure OTG-ZZ-00002, "Reactor Startup," Revision 29; Engineering Technical Procedure ETP-ZZ-ST010, "Low Power Physics Testing Program with Dynamic Rod Worth Measurement," Revision 3; and Engineering Technical Procedure ETP-ZZ-ST006, "Bank Reactivity Worth Measurement," Revision 7, as the bases for acceptability. The inspectors observed the following licensed operator training exercises:

- Licensed operator continued training, "Task Refresher Training for Reactor Startup and Low Power Physics Testing (T61.JT02.8)," observed on November 21, 2002

- Licensed operator continued training, requalification Cycle 02-6, Scenario 1, dated November 21, 2002, "Response to a Loss of Reactor Coolant," observed on December 3, 2002.

- Licensed operator continued training, requalification Cycle 02-6, Scenario 2, dated November 21, 2002, "Response to Steam Generator Tube Rupture," observed on December 3, 2002.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed implementation of the maintenance rule and assessed the effectiveness of maintenance efforts. Specifically, the inspectors reviewed structure and component scoping, characterization, safety significance, performance criteria, and the appropriateness of goals and corrective actions. These aspects of the maintenance rule were reviewed for the following components:

- Steam Generator D Tube Sheet Drain BMV0037

- Residual Heat Removal Pump B Suction from Refueling Water Storage Tank BNHV8812B

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors completed an in-office review and control room inspection of the licensee's risk assessment of two selected emergent maintenance activities. The inspectors compared the licensee's risk assessment and risk management activities against the requirements of 10 CFR 50.65(a)(4); the recommendations of Nuclear Management and Resource Council (NUMARC) 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3; and Engineering Department Procedure EDP-ZZ-01129, "Callaway Plant Risk Assessment," Revision 2. The inspectors also reviewed the effectiveness of the licensee's contingency actions to mitigate increased risk resulting from the degraded equipment. The inspectors evaluated the following risk assessments during the inspection:

- The failure of Train A load shed emergency load sequencer resulting in the inoperability of one off-site power supply and the Train A emergency diesel generator (Callaway Action Request System Number 200206388) on October 16, 2002

- Loss of Train B residual heat removal loop due to a pipe failure (Callaway Action Request System Number 200206766) on October 27, 2002. The inspectors completed a control room and an in-office review on October 28, 2002

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors observed operator performance in coping with six nonroutine events and transients, including one reactor trip from power. The inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and how the operators responded. During the review, the inspectors determined if operator response was in accordance with the response required by procedures and training. The inspectors selected the following events to review operator performance:

- Callaway Action Request System Number 200206766, “Unplanned Loss of Residual Heat Removal Train B During Refueling Mode,” on October 26, 2002.
- Callaway Action Request System Number 200208352, “Unplanned Reactor Trip,” on December 14, 2002
- Callaway Action Request System Number 200208352, “Failure of the Turbine-Driven Auxiliary Feedwater Pump,” on December 14, 2002
- Callaway Action Request System Number 200207348, “Failure to Control the Containment Emergency Personnel Hatch During Mid-Loop Operations,” on November 17, 2002
- Licensee Event Report 2002-002-00, “Motor Driven Auxiliary Feedwater Pump A Inoperable Due to Auxiliary Feedwater Calculation Error,” found on February 6, 2002
- Licensee Event Report 2002-004-00, “Reactor Protection System Actuation While Performing Main Turbine Shell Warming in Mode 4,” on February 13, 2002

b. Findings

.1 Inadequate Design Control Led to an Unplanned Reactor Trip

Introduction

A Green NCV was identified for the failure to adequately control the over temperature-delta temperature (OTDT) delta flux penalty circuit amplifier gain, as prescribed in 10 CFR Part 50, Appendix B, Criteria III. The incorrect amplifier gain resulted in an unplanned reactor trip on December 14, 2002.

Description

Unexpected Bottom Peaked Core Neutron Flux Following Startup

The core axial flux difference was more negative than predicated following reactor restart from Refueling Outage 12. The licensee concluded that the full power flux deviation was approximately 5 percent more negative than expected from an evaluation of the in-core flux map data. The bottom peaked flux profile was driven by axial off-set deviation, a fuel manufacturing deficiency. The axial offset deviation resulted from accumulation of boron dust in the top half of integral fuel burnable absorber fuel assemblies during the manufacturing process. Integral fuel burnable absorber fuel pellets have a thin zirconium-di-boride coating to compensate for the greater fuel reactivity at the beginning of core life. Some of the boron coating became relocated in the top of the fuel rod when the pellets were loaded into the fuel rod. The accumulation of the boron dust at the top of the fuel rod drove a bottom peaked core flux profile during operation. The licensee loaded 96 new fuel assemblies during Refueling Outage 12. All of the new assemblies contained some integral fuel burnable absorber rods which contributed to the axial off-set deviation.

OTDT Core Flux Distribution Penalty Incorrect

The OTDT provides anticipatory reactor protection from conditions that could lead to a departure from nucleate boiling in the core. The OTDT setpoint calculator included an axial core flux distribution penalty to compensate for the potential of high localized fuel rod thermal flux. Axial off-set was defined as the flux measured in the top minus flux measured in the bottom in each core quadrant, as measured by the ex-core detectors. Technical Specification Table 3.3.1-1, “Reactor Trip System Instrumentation,” required the OTDT penalty to be enabled when the circuit detects large core axial off-set (less than a negative 21 or greater than a positive 8 percent at related thermal power). The OTDT circuit also generates a turbine load runback, Control Interlock C-3, when reactor power is within 3 percent of the calculated reactor trip setpoint.

During the past 10 years, the licensee has observed an axial off-set anomaly in the core. The axial off-set anomaly was caused by lithium metaborate building up on the upper half of high powered fuel assemblies. Negative reactivity from the boron in the upper part of the core caused axial power distribution to shift more to the negative at a relatively fast rate. In 1997, the licensee determined that an axial off-set anomaly could potentially change the in-core/ex-core calibration relationship under certain transit conditions. The licensee’s core model predicated insertion of Control Bank D to step 161 from the all rods out position could potentially change the in-core/ex-core calibration relationship up to 4 percent due to control rod shadowing on the peripheral assemblies. Generally, a 2 volt gain is added to the OTDT penalty amplifiers to compensate for the differences in the in-core and ex-core detector outputs over the fuel cycle. The licensee raised the amplifier gain to 3 volts in 1998 following the discovery of a potential nonconservative fuel condition due to an axial off-set anomaly experienced during fuel Cycle 9. With an amplifier gain of 3 volts, the axial core flux penalty began decreasing the OTDT setpoint at a negative 13 percent axial off-set rather than negative 21 percent design value.

Unplanned Reactor Trip

The reactor tripped from a valid OTDT signal on December 14, 2002. Just prior to the trip, the licensee reduced load from 100 percent to 85 percent to remove a damaged condensate pump from service. The condensate pump had developed a significant motor oil leak. The oil leak was the result of a failed compression fitting on the motor cooler water line. The failed fitting resulted in cooling water leaking into the motor housing and displacing oil from the reservoir filler tube. The licensee reduced power by a combination of control rod insertion and boration.

Prior to the reactor transient, axial off-set deviation had driven the core axial off-set to about a negative 11.5 percent. Control rod insertion during power reduction drove the axial off-set slightly more negative. With the 3 volt gain applied to the flux amplifier, the OTDT circuit incorrectly sensed an axial off-set of negative 21 percent and applied the delta flux penalty to the OTDT setpoint calculation. With the penalty, the OTDT setpoint was within 3 percent of the reactor trip setpoint and the circuit initiated a Control Interlock C-3 load runback. The runback resulted in additional control rod insertion. The additional rod insertion resulted in a greater negative flux peak and further lowered the OTDT setpoint, resulting in the reactor trip.

Analysis

The inspectors determined that this condition resulted in a transient initiator that contributed to an initiating event. The initiating event was greater than minor because of the adverse affect on the Reactor Safety Cornerstone objective. If left uncorrected, this finding could reasonably be viewed as a precursor to a more significant event. Based on the "Significance Determination of Reactor Inspection Findings for At-Power Situations," the finding was determined to have very low safety significance. This conclusion was based on:

- The finding not contributing to the likelihood of a primary or secondary system loss of coolant accident initiator,

- The finding not contributing to both the likelihood of a reactor trip and the unavailability of mitigation equipment, and

- The finding not increasing the likelihood of a fire or flood.

Enforcement

Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criteria III, Design Control, states in part, "measures shall be established to assure that applicable regulatory requirements and the design basis, as specified in the license application, for those structures, systems, and components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, the licensee's failure to maintain established design basis requirements for the OTDT axial core flux penalty circuit, and Technical Specification Table 3.3.1-1, "Reactor Trip System Instrumentation," resulted in a premature flux penalty to be enabled, leading to a reactor trip (50-483/0206-01). This finding was entered in to the licensee's corrective action system as Callaway Action Request System Number 200208352.

.2 Turbine-Driven Auxiliary Feedwater Pump Failure

Following the reactor trip, low steam generator water level occurred which resulted in an automatic auxiliary feedwater system actuation. After steam generator levels were restored, the operator secured the turbine-driven auxiliary feedwater pump. A subsequent low steam generator water level condition occurred, resulting in a second automatic auxiliary feedwater system actuation pump initiation. The turbine-driven auxiliary feedwater pump failed to restart due to a turbine overspeed condition.

The licensee determined that the turbine overspeed resulted from governor valve binding. During the turbine start sequence, the governor valve travels from the open position to the throttled position. The turbine overspeed occurred because the governor valve was not able to respond quickly enough to the throttle position due to the binding. The cause of the binding was a combination of misalignment of the governor valve fulcrum support with the bonnet and incorrect governor valve stem size.

The turbine governor valve stem diameter exceeded design specifications. Valve design limited stem diameter variations to between 0.4987 and 0.4995 inches. The valve stem removed from the plant measured between 0.4986 and 0.5008 inches. The excessive stem diameter resulted in stem binding on the valve bonnet carbon bushings. The binding was exacerbated after system heat-up due to thermal expansion of the stem. NRC Information Notice 98-24, “Stem Binding in Turbine Governor Valves in Reactor Core Isolation Cooling and Auxiliary Feedwater Systems,” described the thermal binding of governor valves and the importance of maintaining correct dimensional tolerance between the stem and carbon bushings. The licensee had replaced the governor valve stem during Refueling Outage 12.

Notice of Enforcement Discretion

On December 14, 2002, while in Mode 3, the licensee declared the turbine-driven auxiliary feedwater pump inoperable after the initiation failure on low steam generator water level. The licensee determined that the repair schedule and postmaintenance testing may exceed the Technical Specification 72 hour completion time by up to 48 hours. On December 16, 2002, the licensee requested that the NRC exercise discretion not to enforce compliance with the actions required in Technical Specification 3.7.5, Action C.1, which required the licensee to enter Mode 4 within the following 12 hours.

The NRC granted the licensee’s request for a Notice of Enforcement Discretion based on an evaluation of the proposed compensatory measures and the NRC’s conclusion that extension of the completion time did not have an adverse radiological impact on public health and safety. The NRC issued the Notice of Enforcement Discretion pursuant to the NRC's policy regarding exercise of discretion for an operating facility, set out in Section VII.c of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, and it was effective for the period 5:01 a.m. (CST) December 17 through 5:01 a.m. (CST) December 19, 2002.

The licensee completed turbine-driven auxiliary feedwater pump repair and testing and exited Technical Specification 3.7.5 and the Notice of Enforcement Discretion at 8:14 a.m. on December 17. This issue was considered unresolved pending completion of the NRC’s review of the auxiliary feedwater pump failure root cause and corrective actions (50-483-0206-02).

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors conducted an in-office review and a diesel generator building walkdown of the licensee’s operability evaluation following discovery of a throughwall leak in the Train A emergency diesel generator lube oil cooler (Callaway Action Request System

Number 200206290). The inspectors reviewed the evaluation for technical adequacy and sufficient justification for continued operability of the essential service water system. The inspectors compared the equipment degradation to design basis requirements described in Technical Specification 3.7.8, “Essential Service Water,” and Final Safety Analysis Report Section 9.2.1, “Service Water System.” The inspectors completed the review on October 22, 2002.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed the cumulative effect of operator workarounds on the reliability, availability, and potential for misoperation and on the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors performed an in-office review of the licensee’s November 2002 operator workaround and burden lists and observed the December 2002 plant management operator workaround review meeting.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected six postmaintenance tests that could potentially have affected risk significant systems or components. The inspectors either observed portions of the test or completed an in-office review to verify that each test adequately demonstrated system operability and capability. The inspectors used Technical Specifications, the Final Safety Analysis Report, and ASME Section XI to determine system and component requirements. The inspectors’ review included the following postmaintenance tests:

- Retest R698655, Replacement of essential load shed logic power supply, completed an in-office review on October 30, 2002

- Retest R67539D, Repair of the reactor head vent Valve BBHV8002A, completed an in-office review on November 26, 2002

- Retest R675392A, Removal and re-installation of the containment hydrogen mixing Fan A, completed an in-office review on December 4, 2002

- Retest R620694A, Control rod drive motor-generator set repair, observed testing in the auxiliary building on November 24, 2002, and completed an in-office review on December 4, 2002

- Retest 229225B, Turbine-driven auxiliary feedwater pump turbine, observed testing in the turbine building and the control Building on December 17, 2002, and completed an in-office review on December 24, 2002

- Retest associated with Work Packages W22032 and W22033, Replace flange gasket on residual heat removal Pump B discharge piping, completed an in-office review on December 10, 2002

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated and observed selected refueling outage activities to ensure that the licensee considered plant risk in developing outage schedules, adhered to administrative risk reduction methodologies to control plant configuration, developed appropriate mitigation strategies for losses of key safety functions, and adhered to operating license and Technical Specification requirements.

Outage Plan Review

On September 6, 2002, prior to the outage, the inspectors reviewed the Refueling Outage 12 Risk Analysis to verify that the licensee appropriately considered risk, industry experience, and previous site specific problems. The inspectors also compared the outage plan to licensee mitigation and response strategies for losses of key safety functions. The inspectors compared the outage plan with Administrative Procedure APA-ZZ-00150, “Outage Preparation and Execution,” Revision 12, and Surveillance Report SP02-042, “Assess Risk Management, Preparation and Performance for Cooldown During Mode 3 and Mode 4,” October 29, 2002, during the assessment. The inspectors used NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” December 1991, as a basis for acceptably.

Monitoring of Shutdown Activities

The inspectors observed portions of the reactor cooldown from the control room on October 23, 2002, to verify the licensee did not exceed Technical Specification cooldown limits. The inspectors compared the plant cooldown conditions they observed against Procedure OSP-BB-0007, “Reactor Coolant System Heat up and Cooldown Limitations,” Revision 6.

Licensee Control of Outage Activities

The inspectors attended daily outage meetings and observed the control of outage activities to verify that the licensee maintained defense-in-depth commensurate with the outage risk control plan. The inspectors compared the licensee's evaluation of emergent work risk with Engineering Department Procedure EDP-ZZ-1129, "Callaway Plant Risk Assessment," Revision 0; NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management"; and Surveillance Report SP02-042, "Refuel 12 Plant Operations," October 29, 2002.

- **Clearance Activities**

  The inspectors conducted a containment building walkdown of Workman's Protection Assurance System 44045, "Isolation of Reactor Coolant Loop 1 Hot Leg," on November 5, 2002, and Callaway Modification Package 01-1019, "Replacement of Reactor Coolant Check Valve BB8949A," to verify that tags were properly hung and removed and that associated equipment was appropriately configured to support the function of the clearance. The inspectors also conducted a turbine building and control room walkdown of Workman's Protection Assurance System 44987, "Installation of Main Generator Grounds, Back-Feed Plant Power Through Main Transformers," on November 8, 2002.

- **Reactor Coolant System Instrumentation**

  The inspectors conducted plant walkdowns and performed control room observations to verify that reactor coolant system pressure, level, and temperature instruments were installed and configured to provide accurate indication. The inspectors also reviewed the status and configuration of electrical systems to verify that Technical Specification requirements and the licensee's outage risk control plan requirements were met.

- **Electrical Power**

  The inspectors periodically reviewed the status and configuration of electrical systems to verify that the licensee met Technical Specification requirements and the outage risk control plan. The inspectors also observed switchyard activities, on November 8, 2002, to verify they were controlled commensurate with safety and are consistent with the licensee's outage risk control plan assumptions.

- **Decay Heat Removal System Monitoring**

  The inspectors observed the licensee's decay monitoring system, from the control room, to verify the system was properly functioning. The inspectors observed single phase natural circulation when the licensee relied on the steam generators to provide a backup means of decay heat removal. The inspectors also observed the performance of Operations Surveillance Procedure OSP-ZZ-0001, "Control Room Shift and Daily Log Readings and Channel Checks," Revision 39, on October 24 and 30 and November 12, 2002.

- **Spent Fuel Pool Cooling System Operation**

  The inspectors reviewed planned outage activities to verify that outage work did not impact the ability of the operations staff to operate the spent fuel pool cooling system during and after core offload. The inspectors also confirmed, during periods of core offload, that adequate spent fuel pool cooling was being maintained in accordance with Technical Specifications.

- **Inventory Control**

  The inspectors observed the reactor cavity flood-up from the control room and the containment building on October 28, 2002, to verify that the flow paths, configurations, and alternative means for inventory addition were consistent with the outage risk plan. The inspectors also observed the licensee’s actions following an unplanned loss of residual heat removal (Callaway Action Request System Number 200206766) on October 26, 2002, to verify that there were adequate controls in place to prevent inventory loss or other conditions which had the potential to cause a loss of inventory.

- **Reactivity Control**

  The inspectors reviewed the licensee’s outage reactivity controls to verify that the Technical Specification requirements were met. The inspectors performed auxiliary building and control room walkdowns to verify that the licensee maintained required boron injection flow paths. The inspectors also reviewed outage activities that could cause unexpected reactivity changes.

- **Containment Closure**

  The inspectors observed midloop and reduced reactor coolant inventory operations and conducted containment building walkdowns on November 17, 2002, to verify that the licensee controlled containment penetrations in accordance with the administrative controls and Technical Specifications. The inspectors compared plant observations with Administrative Procedure APA-ZZ-00150, “Outage Preparation and Execution,” Revision 12; Operations Surveillance Procedure OSP-ZZ-00001, “Control Room Shift and Daily Log Readings and Channel Checks,” Revision 39; and Operations Surveillance Procedure OSP-SF-00003, “Pre-Core Alteration Verifications,” Revision 12. The inspectors also performed an in-office review of Callaway Action Request System Number 200206835, “Containment Equipment Hatch Status for Upper Internals Lift,” October 28, 2002, and Surveillance Report SP02-038, “Reduced Inventory Controls,” October 25, 2002.

Reduced Inventory and Midloop Conditions

The inspectors reviewed the licensee’s commitments from Generic Letter 88-17, “Loss of Decay Heat Removal,” and confirmed, by sampling, that they were still in place and adequate. The inspectors observed reduced inventory operations from

November 14-17, 2002, from the control room and the containment building. The inspectors assessed the effect of distractions, due to unexpected plant conditions or emergent activities, on the operators' abilities to maintain the required reactor vessel level during midloop operations.

Refueling Activities

The inspectors observed fuel handing activities to verify operations were performed in accordance with Technical Specifications and approved procedures. The inspectors observed fuel handing coordination from the control room to verify that the licensee tracked the location of fuel assemblies from core offload through core reload.

Monitoring of Heat up and Startup Activities

The inspectors observed reactor heat-up and reactor physics testing, November 24-26, 2002, to verify that Technical Specifications, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following six surveillance tests to ensure that the systems tested were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors verified that the following surveillance tests met Technical Specifications, ASME Section XI test requirements, the Final Safety Analysis Report, and licensee procedural requirements:

- Instrumentation Surveillance Procedure ISP-SA-2413B, "Diesel Generator and Sequencer Testing (Train B)," Revision 14, observed on November 18, 2002
- Operations Surveillance Procedure OSP-NE-0024B, "Emergency Diesel Generator B, 24 Hour Run and Hot Start," Revision 6, observed on November 12, 2002
- Instrumentation Surveillance Procedure ISP-SA-2413B, "Diesel Generator and Sequencer Testing (Train B)," Revision 14, observed November 19, 2002
- Mechanical Surveillance Procedure MSM-EN-QW001, "Containment Spray System Nozzle Flow Test," Revision 8, observed on November 15, 2002, and an in-office review completed on December 20, 2002

- Engineering Test Procedure ETA-AL-ST016, “A Train Auxiliary Feedwater Vibration Testing,” Revision 0, completed an in-office review on October 30, 2002

- Operations Surveillance Procedure OSP-ZZ-00001, “Control Room Shift and Daily Readings and Channel Check, “Revision 39, observed November 11, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors completed a review of four temporary modifications during the inspection period. The inspectors compared each temporary modification package against the requirements established in Administrative Procedure APA-ZZ-00605, “Temporary System Modifications,” Revision 14, and system requirements contained in Technical Specifications and the Final Safety Analysis Report.

- Temporary Modification 02-0017, Change of fuel transfer system load setpoint, installed November 5, 2002. The inspectors completed in-office and control room reviews on November 26, 2002.

- Procedurally Controlled Temporary Modification/Engineering Technical Procedure ETP-SE-ST-001, “Nuclear Instrumentation Startup Testing,” Attachment 5, Reset of the nuclear instrumentation high trip setpoint. The inspectors completed an in-office review of the supporting safety evaluation and a control room walkdown of the installation on November 26, 2002.

- Procedurally Controlled Temporary Modification/Engineering Technical Procedure ETP-SE-ST-003, “Pre-Critical Alignment and Hookup of Advanced Digital Reactivity Computer,” Attachment 4, “Installation and Removal.” The inspectors completed an in-office review of the supporting safety evaluation and a control building walkdown of the installation on November 24, 2002.

- Procedurally Controlled Temporary Modification/Engineering Technical Procedure ETP-SE-ST-003, “Pre-Critical Alignment and Hookup of Advanced Digital Reactivity Computer,” Attachment 3, “Adjustment of Control and Shutdown Back Rod Speeds.” The inspectors completed an in-office review of the safety evaluation and a control building walkdown of the installation on November 24, 2002.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during normal operations. The inspectors also conducted plant walkdowns within the radiological controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Quality Assurance Audits AP01-002, AP01-003, and AP02-001 pertaining to access controls and radiological work practices
- Area posting and other controls for airborne radioactivity areas, radiation areas, high radiation areas, locked high radiation areas, and very high radiation areas
- Access controls, radiological surveys, and radiation work permits for the following three significant high dose work jobs: leak identification and repair in letdown cubicle (110520LEAK), repair of Valve ECV0995 leak-by (W210714), and reactor coolant system filter replacement (G681744022)
- Dosimetry placement when work involved a significant dose gradient
- Controls involved with the storage of highly radioactive items in the spent fuel pool
- A summary of access controls and high radiation area work practice related corrective action documents written since August 2001 and selected specific examples: Callaway Action Request System Numbers 20020176, 20021061, 20022511, 20022928, 20023323, 20023737, 20023830, 20024481, 20024820, and 20026122

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Reactor Safety Cornerstones

a. Inspection Scope

The inspectors sampled licensee submittals for the five performance indicators listed below for the period from October 2001 through September 2002. The inspectors used performance indicator definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 1, to verify the accuracy of the data reported during the period.

- Safety System Unavailability - High Pressure Safety Injection System
- Safety System Unavailability - Residual Heat Removal System
- Safety System Functional Failures
- Initiating Events - Unplanned Scrams
- Initiating Events - Scrams with Loss of Normal Heat Sink

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed corrective action program records for Technical Specification required locked high radiation areas (as defined in Technical Specification 5.7.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned exposure occurrences (as defined in NEI 99-02) for the past 12 months to confirm that these occurrences were properly recorded as performance indicators radiological controlled access area entries with exposures greater than 100 millirems were reviewed, and selected examples were examined to determine whether they were within the dose projections of the governing radiation work permits. Whole-body counts or dose estimates were reviewed if the radiation worker received a committed effective dose equivalent of more than 100 millirems.

b. Findings

No findings of significance were identified.

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors reviewed radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented during the past four quarters to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds (as defined in NEI 99-02).

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors performed detailed in-office reviews and walkdowns of plant equipment related to seven conditions adverse to quality reports. The reports were reviewed to ensure that the full extent of the issues was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of the licensee's corrective action program, Administrative Procedure APA-ZZ-00500, "Corrective Action Program," Revision 21, and 10 CFR Part 50, Appendix B. The inspectors review included:

- Callaway Action Request System Number 200207498, Failure to replace two residual heat removal flow elements during maintenance
- Callaway Action Request System Number 200207880, Unplanned reactor protection system actuation during testing
- Callaway Action Request System Number 200206835, Loss of electrical supply Bus NB02 due to the failure to follow procedure
- Callaway Action Request System Number 200206811, Containment penetrations not administratively controlled during core alterations
- Callaway Action Request System Number 200206870, Failure to administratively control hoses passed through the emergency personnel hatch during core alterations
- Callaway Action Request System Number 200207137, Foreign material inadvertently left in the emergency diesel generator following maintenance
- Callaway Action Request System Number 200207358, Foreign material discovered in the containment coolers

b. Findings

No findings of significance were identified.

4OA5 Other

.1 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (Temporary Instruction 2515/145)

a. Inspection Scope

From October 31 to December 4, 2002, the inspectors performed NRC Inspection Manual Temporary Instruction 2515/145 during Refueling Outage 12. They reviewed the licensee's inspection plan and the NRR assessment in response to NRC Bulletin 2001-01. The inspectors noted that Callaway was considered a Low-Susceptible Plant (Bin 4) according to the bulletin. The inspectors noted that NRC Bulletin 2001-01 recommended a 100 percent effective visual examination of the surface of the reactor vessel head and the annulus area around each penetration nozzle. The licensee's methodology employed the use of small (less than 1 foot in length) remotely controlled robots equipped with video cameras. These robotic units crawled over the exterior surface of the reactor vessel head and relayed visual information to analysts for review. The licensee also performed a visual inspection of the outer periphery of the reactor vessel (where the physical clearances were too tight to allow robotic access); no signs of boric acid leakage or trails were identified.

Through discussions with the licensee, and conferences with NRR, the inspectors assessed the validity of this methodology to meet the intent of NRC Bulletin 2001-01. The inspectors reviewed the demonstration of these methods and their ability to determine flaws in j-groove welds and base metals associated with primary water stress corrosion cracking. The inspectors conducted interviews with plant engineers and Westinghouse contractors to determine their training, background, and expertise in conducting and analyzing these examinations, including questions regarding the methods used to ensure complete documentation and review of all findings. The inspectors observed the equipment operation and a sample of the actual nozzle testing. They also observed Westinghouse contractors perform analyses of the visual data for multiple nozzles.

b. Findings

No findings of significance were identified.

.2 Review of Evaluation and Accreditation Reports by the Institute of Nuclear Power Operations

The Institute of Nuclear Power Operations conducted an evaluation of site activities at the Callaway Plant. The inspectors completed a review of the August 5, 2002, report detailing the assessment of the Callaway Plant.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. R. Affolter, Vice President, Nuclear, and other members of licensee management at the conclusion of the access control to radiologically significant areas inspection on October 25, 2002.

The inspectors presented the inspection results to Mr. R. Affolter, Vice President, Nuclear, and other management representatives at the conclusion of the inservice inspection on October 31, 2002.

The inspectors presented the inspection results to Mr. W. Witt, Plant Manager, and other management representatives at the conclusion of the steam generator tube inservice inspection on November 8, 2002.

The inspectors presented the inspection results to Mr. R. Affolter, Vice President, Nuclear, and other members of licensee management at the conclusion of the resident inspection on January 6, 2003.

In each case, the inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Affolter, Vice President, Nuclear
M. Evans, Manager, Nuclear Engineering
T. Herrmann, Superintendent, Engineering
G. Hughes, Acting Manager, Quality Assurance
B. Huhmann, Supervising Engineer, Steam Generator Replacement
J. Laux, Manager, Operations Support
V. McGaffic, Superintendent, Performance Improvement
M. Reidmeyer, Supervisor, Regulatory Affairs
R. Roselius, Superintendent, Health Physics
E. Thornton, Engineering Evaluator, Quality Assurance
W. Witt, Plant Manager

ITEMS OPENED AND CLOSED

Opened

50-483/0206-01	NCV	Failure to maintain established design basis requirements for the OTDT axial core flux penalty circuit (1R14.b.1)
50-483/0206-02	URI	Inoperable turbine-driven auxiliary feedwater pump (1R14.b.2)

Closed

50-483/0206-01	NCV	Failure to maintain established design basis requirements for the OTDT axial core flux penalty circuit (1R14.b.1)

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Callaway Action Requests

200200094
200200316
200202197
200202256
200203891
200204064
200204953
200206766
200206839
200206842
200206976
200207229
200207498
20028280
200208312

Drawings:

NUMBER	TITLE	REVISION
M-22EJ01(Q)	Piping and Instrumentation Diagram Residual Heat Removal System	44
M-23EJ04(Q)	Piping Isometric Residual Heat Removal System Reactor Building	13
M-23EJ07(Q)	Small Piping Isometric Residual Heat Removal System Reactor Building	5

Operability Determinations

200206766
200206839
200206976

Procedures

NUMBER	TITLE	REVISION OR DATE
MTW-ZZ-WP514	Welding of P-8 Materials	11
83A6106	Liquid Penetrant Examination Procedure	0
83A6216	Ultrasonic Examination for Ferritic Piping Welds	7
83A6226	Ultrasonic Examination Procedure for Austenitic Piping Welds	6
ETP-BB-01309	Steam Generator Eddy Current Testing Acquisition and Analysis Guidelines	13
54-ISI-24-27	Written Practice for Personnel Qualification in Eddy Current Examination	1/9/01
ETP-BB-01338	Remote Rolled Plugging Utilizing LAN SAP Box	4
ETP-ZZ-01300	Multi-Frequency Eddy Current Examination	11

Procedures

NUMBER	TITLE	REVISION OR DATE
ETSS 1	Examination Techniques Specification Sheet	1
ETSS 2	Examination Techniques Specification Sheet	0
ETSS 4	Examination Techniques Specification Sheet	0
OTS-ZZ-00007	Plant Cold Weather	6
OTN-BB-00002	Reactor Coolant System Draining	24
OTO-EJ-00001	Loss of RHR Flow	13
OTS-BB-00007	Dynamic Fill and Vent of the RCS	1
OTO-EJ-00002	Loss of RHR Due to Heavy Load Drop in Containment	1
OTN-BB-00001	Reactor Coolant System	16
EDP-ZZ-01004	Boric Acid Corrosion Inspection Program	0
FPP-ZZ-00002	Fire Preplan Manual (Attachments 1 & 2)	2
FPP-ZZ-00003	Fire Preplan Manual (Attachments 2, 3 & 4)	3

Request for Resolution

22409, Evaluate Repair of Leak Near EJV0179, Revision A
13493, Remove Insulation on Pressurizer Safety Loop Seals, Revision A

Work Document

W223102
S675732

Licensee Event Reports

2000-006-00	2001-001-00	2001-002-00
2001-003-00	2001-004-00	2001-005-00
2002-001-00	2002-002-00	2002-003-00
2002-004-00	2002-005-00	2002-006-00

2002-007-00	2002-008-00	2002-009-00
2002-010-00	2002-011-00	

Miscellaneous Documents

Technical Specification 5.5.9

Event Notification 39345

Policy Document UEND-Strategy-02, “Steam Generator Strategic Plan for Callaway”

Nuclear Energy Institute NEI 97-06, “Steam Generator Program Guidelines” Revision 1

EPRI Guidance Document TR-107620, “Steam Generator In Situ Pressure Test Guidelines,” Revision 1

Document 51-5019993-00, “Callaway Degradation Assessment, Refuel 12", Revision 12

Document R-4136-00-3, “Callaway Cycle 12 Operational Assessment,” Revision 0

Licensee Equipment out of Service Log reports

Licensee performance indicator worksheets

NRC Web Site Performance Indicator data

Performance indicator summary reports

Selected NRC inspection reports