....September 15, 1999

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1700 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: NRC INSPECTION REPORT 50-254/99017(DRS); 50-265/99017(DRS)

Dear Mr. Kingsley:

On August 27, 1999, the NRC completed the pilot baseline annual inspection of Changes to the Safety Analysis Report and the biennial Permanent Plant Modifications inspection at your Quad Cities Nuclear Station. The results of these inspections were discussed on August 27, 1999, with Mr. J. Dimette and other members of your staff. The enclosed report presents the results of these inspections.

The inspections were an examination of activities conducted under your license as they relate to changes to the Updated Final Safety Analysis Report under the provisions of 10 CFR 50.59 and changes to the facility via permanent plant modifications to verify compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

One minor issue regarding the failure to retain a post modification test was identified. This issue was categorized as Green. In addition, several previously identified issues were evaluated and one additional example of a previous violation was identified regarding updating of the Updated Final Safety Analysis Report. Enforcement discretion was applied to the previous violation and no additional actions are necessary for this new example. However, if you contest the addition to the violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Original /s/ John M. Jacobson

John M. Jacobson, Chief
Mechanical Engineering Branch

Docket Nos. 50-254; 50-265
License Nos. DPR-29; DPR-30

Enclosure: Inspection Report 50-254/99017(DRS);
50-265/99017(DRS)

cc w/encl: D. Helwig, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President, Nuclear Operations
H. Stanley, Vice President, Nuclear Operations
R. Krich, Vice President, Regulatory Services
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J. Dimmette, Jr., Site Vice President
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C. Peterson, Regulatory Affairs Manager
M. Aguilar, Assistant Attorney General
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
W. Leech, Manager of Nuclear
MidAmerican Energy Company

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-254; 50-265
License Nos:	DPR-29; DPR-30
Report No:	50-254/99017(DRS); 50-265/99017(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Quad Cities Nuclear Power Station Units 1 and 2
Location:	22710 206th Avenue North Cordova, IL 61242
Dates:	August 23 - August 27, 1999
Inspectors:	David Butler, Reactor Inspector Patricia Lougheed, Reactor Inspector Roger Mendez, Reactor Inspector
Approved by:	John M. Jacobson, Chief, Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Quad Cities Nuclear Power Station, Units 1 & 2
NRC Inspection Report 50-254/99017(DRS); 50-265/99017(DRS)

This report covers the pilot baseline inspections for the annual review of changes to the safety analysis report and the biennial review permanent plant modifications.

Inspection findings were assessed according to potential risk significance, and were assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while not necessarily desirable, represent little risk to safety. WHITE findings would indicate issues with some increased risk to safety, and which may require additional NRC inspections. YELLOW findings would be indicative of more serious issues with higher potential risk to safe performance and would require the NRC to take additional actions. RED findings represent an unacceptable loss of margin to safety and would result in the NRC taking significant actions that could include ordering the plant shut down. No individual finding by itself would be indicative of either acceptable or unacceptable performance. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

Cornerstone: Mitigating Systems

Green: The inspectors identified that the Unit 1 post modification test for a design change package on the fuel transfer pump was not retained by the licensee. The licensee had retained the Unit 2 test and had signature evidence that the Unit 1 test was performed.

Cornerstone: Barrier Integrity

No findings were identified in this area.

Report Details

1. **REACTOR SAFETY**
Cornerstones: Mitigating Systems and Barrier Integrity

1R02 Changes to License Conditions and Safety Analysis Report (IP 71111, Attachment 2)

.1 Review of 50.59s Evaluations for Changes to the Safety Analysis Report

a. Inspection Scope

The inspectors reviewed five evaluations done pursuant to 10 CFR 50.59, one of which pertained to the barrier integrity cornerstone. All five evaluations related to changes to the updated final safety analysis report. The inspectors also reviewed three changes to the updated final safety analysis report where the licensee had determined that a 50.59 evaluation was not necessary. In regard to the three changes where no 50.59 evaluation was performed, the inspectors verified that the changes were minor editorial clarifications that did not meet the threshold of a "change to the facility as described in the safety analysis report." For the 50.59 evaluations, the inspectors confirmed that prior NRC approval was not required for any of the changes.

b. Observations and Findings

No findings were identified in this area.

.2 (Closed) URI 50-254/98201-18; 50-265/98201-18: Updated Final Safety Analysis Report Discrepancies. This unresolved item was previously closed in Inspection Report 98019 with the exception of one item dealing with Tables 8.3-2 and 8.3-3 regarding emergency diesel generator loading. The licensee acknowledged that the information in Tables 8.3-2 and 8.3-3 had changed, that the emergency diesel generator loading information was design basis information, and that an updated final safety analysis report revision was necessary. The cognizant regulatory assurance engineer revised the unresolved item action tracking item (000156) to track this issue. The licensee also reviewed other Section 8 updated final safety analysis report tables to determine if those tables contained design basis information. In one case, the licensee discovered that the information presented in the updated final safety analysis report table was not available in the load tracking database, despite a note that referenced the reader to that database. Therefore, the licensee expanded the action tracking item to address whether these tables needed to remain in the updated final safety analysis report and whether the reference to the load tracking database was necessary in all cases. The inspectors reviewed the current load tracking database and determined that the emergency diesel generators loadings were within their design basis values. Therefore, the inspectors had no further technical questions.

10 CFR 50.34(b)(2) requires, in part, that the final safety analysis report contain a description and analysis of the structures, systems and components of the facility, with emphasis upon performance requirements, the bases upon which the requirements were established, and the evaluations required to show that the safety functions will be accomplished. 10 CFR 50.71(e) requires, in part, that each licensee periodically update the final safety analysis report to assure that the information included in the final safety analysis report contains the latest material developed.

The loading on the emergency diesel generators following a loss of coolant accident and/or a loss of offsite power are performance requirements that are necessary to understand the emergency diesel generator system design and safety evaluation. Therefore, they are required to be in the updated final safety analysis report by 10 CFR 50.34(b)(2). The failure to update the design performance requirements for the emergency diesel generator loadings is a violation of 10 CFR 50.71(e). However, this item was identified as part of a larger unresolved item for which enforcement discretion was already granted (VIO 50-245/264-98201-07). Therefore, it will be considered another example of that violation and no separate enforcement action will be taken.

.3 (Closed) Unresolved Item (URI) 50-254/97013-02; 50-265/97013-02: Updated Final Safety Analysis Report Discrepancy Regarding the Residual Heat Removal Service Water Pumps. In 1997, the NRC identified a discrepancy among the pump surveillance test requirements, the design basis document and the updated final safety analysis report regarding the required residual heat removal service water pressure. The inspectors confirmed that the pumps were capable of meeting their design function of removing heat from the residual heat removal system following a design basis accident. Therefore, the inspectors had no further technical questions.

The inspectors also reviewed the approved updated final safety analysis report change (UFSAR-97-R5-016) and associated 10 CFR 50.59 evaluation. The actual change was to add a footnote denoting the information as being the "original manufacturer's specification of the size of the pumps chosen." The wording implied that the pumps were no longer capable of meeting their original design specification or function. Additionally, the updated final safety analysis report did not address the pumps functional requirement (to provide water to the residual heat removal heat exchanger at a specified flow and pressure) which was different than the pump design information. Therefore, the licensee revised the planned corrective actions for problem identification form Q1999-02345 to ensure that the design basis of the pumps was adequately captured in the updated final safety analysis report. This item is closed.

1R17 Permanent Plant Modifications (IP 71111, Attachment 17)

.1 Review of Recent Plant Modifications

a. Inspection Scope

The inspectors reviewed seven plant modifications which were installed since August 1998. The packages were chosen based upon their affecting systems that had high Maintenance Rule safety significance or high risk significance in the licensee's Individual Plant Evaluation. Five of the modifications involved changes to mitigating systems, while the last two affected barrier integrity. The inspectors reviewed the modifications to confirm that the changes did not affect any systems' safety function. Design and testing aspects were verified to ensure the functionality of the modification, it's associated system, and any support systems. Walkdowns were conducted to ensure proper installation of the modifications.

b. Observations and Findings

The inspectors identified that the Unit 1 post modification test for a design change package on the fuel transfer pump was not retained by the licensee. Retention of the post modification test would have provided evidence that the test was adequately performed. The licensee had retained the Unit 2 test and signature evidence that the Unit 1 test was performed.

The post modification test to demonstrate operability of the fuel transfer pump was outlined in the design change acceptance testing summary and was to be performed in accordance with surveillance procedure QCOP 4100-16, "Manually Filling the Design Fire Pump Day Tank," Revision 2. However, the procedure was not considered a quality document and, therefore, the completed test was not retained for the life of the plant when the design change package was microfilmed. The licensee subsequently produced a copy of the signed cover sheet and the work request form that required the test. In addition, the system engineer stated that the test was adequately performed. This assured the inspectors that the test was performed. The licensee committed to review the requirements for retaining post modification test records for surveillances that were not required to be retained.

The inspectors verified that there were adequate assurances that the fuel transfer pump would operate as modified. The inspectors performed a Phase I screening of this issue under the significance determination process and the issue screened out as "Green".

4 OTHER ACTIVITIES

4OA4 Other (IP 93902)

.1 (Closed) URI 50-254/97022-02; 50-265/97022-02: Breaker Coordination Issues. The inspectors reviewed the Station's position paper (White Paper) prepared for this item and the 250 Vdc system licensing basis. The White Paper provided additional information demonstrating that partial breaker coordination existed. The lack of breaker coordination occurred in the breakers instantaneous tripping region. This mis-coordination would only result from faults that occurred at the breaker load side terminals or by a cable fault within close proximity of the load breaker. Since cables are highly reliable and a failure at the breaker load terminals is highly unlikely, the most likely fault condition would be at the load. Due to cable length, load fault currents would be limited allowing the load breaker to clear the fault without tripping the upstream feed breaker. In addition, the licensee indicated that breaker coordination was a design consideration and that they attempted to optimize breaker coordination during the design process. Therefore, the inspectors determined that there was no technical concern with the licensee's approach. This item is considered closed.

.2 (Closed) Violation 50-254/98019-04; 50-265/98019-04: Inadequate Corrective Action. In Inspection Report 50-254/265-98019, the inspectors noted that the licensee had taken adequate corrective actions to the violation and that no response was necessary. Therefore, this violation is closed.

.3 (Closed) Violation 50-254/98019-05; 50-265/98019-05: Inadequate Design Control. In Inspection Report 50-254/265-98019, the inspectors noted that the licensee had taken adequate corrective actions to the violation and that no response was necessary. Therefore, this violation is closed.

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management in an exit meeting on August 27, 1999. The licensee acknowledged the information and findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Barnes, Station Manager
J. Dimmette, Site Vice President
M. McDonald, Operations Manager
C. Peterson, Regulatory Assurance
D. Wozniak, Engineering Manager

NRC

J. Caldwell, Deputy Regional Administrator, Region III
L. Collins, Resident Inspector
J. Jacobson, Chief, Mechanical Engineering Branch, DRS
S. Reynolds, Deputy Division Director, Division of Reactor Safety
M. Ring, Chief, Branch 1, Division of Reactor Projects

INSPECTION PROCEDURES USED

IP 71111.02 (draft)	Changes to License Conditions and Safety Analysis Report
IP 71111.17 (draft)	Permanent Plant Modifications
IP 93902	Followup - Engineering

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Closed (254/265)

97013-02	URI	Updated Final Safety Analysis Report Discrepancy for Residual Heat Removal Service Water Pumps: Two Issues
97022-02	URI	Breaker Coordination Issues
98201-18	URI	Updated Final Safety Analysis Report Discrepancies
98019-04	VIO	Inadequate Corrective Action
98019-05	VIO	Inadequate Design Control

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort.

Calculations

ATD-0057	Evaluation of Bore Diameter of Unit 2 Restricting Orifices RO 2-3924 and 2-3925, Revision 0
QC-270-C-024	Cable Tray Loading Calculation, Revision 7
QDC-5200-S-0732	Seismic Qualification of CP 2940 Switch Addition to Diesel Generator Panel, Revision 0
QDC-6600-S-0722	Seismic Qualifications of TD2 Time Delay Relay and Mounting of Enclosure and Conduit, Revision 0

Drawings

4E-1350C	Schematic Control Diagram of the Diesel Fuel Oil Transfer Pump 1 and 2 Feed Controls, Revisions K & M
4E-1351A	Schematic Control Diagram of the Engine Control and Generator Excitation for the Standby Diesel Generator 1/2, Revisions AG, AK, & AL
4E-1358A	Schematic Diagram Electro-Hydraulic Control System, Revision K
4E-2358B	EHC Alarm and Trip Schematic, Revision K
4E-2629	Turbine EHC Pressure Switches, Revision K
4E-2637	EHC Cabinet No. 902-31 Alarm and Trip Wiring Diagram, Revision D
4E-2655J	Wiring Diagram of the 4160V Switchgear Bus 23-1 Cubicle 8, Revision J
4E-6820H	EHC Power-Load Unbalance Demodulator Schematic, Revision G
4E-6823M	EHC Power-Load Unbalance Circuit Schematic, Revision H
4E-6872F	Wiring Diagram of the Exterior Cable Tray Layout Station Blackout Sections and Details, Revision C
4E-7871C	Wiring Diagram of the Station Blackout 4160 Switchgear Bus 71 Cubicle 3, Revision A
M-4A-1, 2,3,& 4	Environmental Zone Maps
M-36	Diagram of Core Spray Piping, Revision AU
M-37	Diagram of Residual Heat Removal Service Water Piping, Revision AP
M-39-1,2,3	Diagram of Residual Heat Removal Piping, Revisions BC, AW, & B
M-43	Diagram of Reactor Building Equipment Drains, Revision BJ
M-46-1,2	Diagram of High Pressure Coolant Injection Piping, Revisions BN & F
M-47-1	Diagram of Reactor Water Cleanup Piping, Revision --
M-50-1	Diagram of Reactor Core Isolation Cooling Piping, Revision BA
M-75-2	Diagram of Reactor Building Closed Cooling Water Piping, Revision B
M-77-2	Diagram of Nuclear Boiler and Reactor Recirculating Piping, Revision AN
M-725-1,2,3	Diagram of Control Room Heating, Ventilation, and Air-Conditioning System, Revisions K, K, & E
M-2022	Turbine Control Diagram Units 1 and 2, Revision A

Electrical Standards

EM-29105	600 Volt Power Cable for Nuclear Generating Stations, September 1,1992
EM-29115	600 Volt Control Cable for Nuclear Generating Stations, June 1, 1994
EM-29116	5KV Ethylene Propylene Insulated Chlorosulfonated Polyethylene Jacketed Power Cable for Stations and Substations, December 14, 1990
EQ-GEN029	3M Scotch Tape Splices, Revision 4

N-C-0008	Cable Pulling Guidelines, Revision 5
N-EM-0035	Cable Standards, Revision 6
N-EM-0048	Low Voltage Tapes, Revision 2

Engineering Requests

ER9805134	Revise General Electric Specification (GEK) No.11367C Steam Turbine Instruction Manual for Electrical Alarm and Trip System (Pressure Switches PS-101A and 101B)
ER9900536	Revise Procedure QCIPM 5610-31, Revision 3, “EHC Low Hydraulic Pressure Turbine Trip Functional Test”
ER9805001	Revise General Electric Specification GEK No. 11367C Steam Turbine Instruction Manual for Electrical Alarm and Trip System (High Head Temperature Bypass above 30 Percent Pressure)
ER9804426	Revise Procedure No. QOA 5600-03, Revision 5, “Turbine Hood Spray Regulation Valve Failure”
ER9804425	Revise Procedure No. QOA 5600-04, Revision 12, “Loss of Turbine Generator”

General Electric (GE) Specifications

GEH 3626	Pressure Switch Model CR127A
GEK 11365A	ElectroHydraulic Contros
GEK 11354A	Power/Load Unbalance Circuit and Relays

GE Technical Service Letters

1212-2	Plant SCRAM Frequency Reduction Features for BWR and PWR Nuclear Turbines with MKI and MKII EHC Controls, January 27, 1997

Miscellaneous

Control Room Habitability Study, Revision 2, 6/14/1982
Licensed Operator Continuing Training, “Mods and Lessons Learned 99-1," January 12, 1999

Modifications

DCP 9700346	Modify Turbine Trip Logic (EHC Low Power) to Reduce the Probability of a Spurious Trip
DCP 9700351	Enlarge Bore Sizes of Reducing Orifices 2-3924 and 2-3925
DCP 9700366	Modify Turbine Trip Logic from Low Pressure Turbine Exhaust Hood High Temperature to Bypass the Trip Above 30% Turbine/Generator Power
DCP 9800226	Replace Emergency Diesel Generator Time Delay Relay TD2
DCP 9800233	Splice Station Blackout Diesel Control Cables and Reroute Power Cable to Switchgear 23-1
DCP 9800235	Modify the Emergency Diesel Generator Fuel Transfer Pump Logic
DCP 9900009	Modify the 24 Volt Circuit Powering the Unit 2 Scram Discharge Volume Instruments

Nuclear Design Information Transmittals

QDC-99-071	Design Input - ECCS Room Coolers

Part Evaluations

NEP-18-04	Equipment Dynamic Qualification Test Review of Time Delay Relays

Problem Identification Forms

Q1997-04066	High Pressure Coolant Injection High Steam Flow Switch Logic and updated final safety analysis report Wording, 10/26/1997
Q1999-02345	Residual Heat Removal Service Water Pump Testing and Safety Margin of Pumps, 7/19/1999

Procedures

ECTP-19	Control Circuits, Revision 2
ECTP 24	Operational Analysis Department Electrical Construction Test Procedure for Modifications at Nuclear Stations, Revision 2
NEP 04-01	Plant Modifications, Revision 6
NEP-04-05	Design Change Acceptance Testing Criteria, Revision 0
NEP 14-03	Control and Tracking of Electrical Load Changes, Revision 1
NSWP E-02	Electrical Cable Termination and Inspection, Revision 5-1
PMID 154086	VC2A15 (A15-VC2) Voltage Comparator Calibration
QCAP 0200-15	High Risk Activity Mitigation Plan, Revision 9
QCAP 1100-13	Procedure Field Change Request, Revision 8
QCEM 700-1	Cable Pulling Procedure, Revision 0
QCEPM 0700-18	Calibration of Diesel Generator Time Delay Relays, Revision 7
QCOP 4100-16	Manually Filling the Diesel Fire Pump Day Tank, Revision 2
QCOP 5750-09	Control Room Ventilation System, Revision 15
QCOS 0010-07	Equipment External Leak Test, Revision 1
QCOS-0201-12	Class One ASME Section XI Post-replacement Pressure Test at Power Operation, Revision 0 (also completed procedure 3/1/1999)
QCOS 1000-04	Quarterly Residual Heat Removal Service Water Pump Operability Test, Revision 12
QCOS 5750-09	ECCS Room and DGCWP Cubicle Cooler Monthly Surveillance, Revision 15
QCOS 6600-03	Diesel Fuel Oil Transfer Pump Monthly Operability, Revision 6
QCOS 6620-01	SBO DG 1(2) Quarterly Load Test, Revision 3
QCIP 0100-05	Instrument Maintenance Department Administrative Drift Limits Guidance, Revision 2
QCIPM 5610-30	EHC Low Hydraulic Pressure Turbine Trip Functional Test, Revision 3
QCIPM 5610-33	Turbine Exhaust hood High Temperature Turbine Trip Functional Test, Revision 2
QCTS 0220-02	Unit 1 and 2 24/48 Vdc Battery Performance Test, Revision 4
QCTS 0220-05	Unit 1 and 2 24/48 Vdc A Battery Service Test, Revision 2
QCTS 0220-08	Unit 1 and 2 24/48 Vdc B Battery Service Test, Revision 1
QIP 0100-18	Refuel Outage Balance of Plant Calibration Schedule, Revision 12
QOA 900-7 C-4	901-7 (902-7) Row C Annunciator Procedures, Revision 2
QOS 0005-S01	Operations Department Weekly Summary of Daily Surveillance, Revision 93
SPP-VT-2-1	VT-2 Visual Inspection Performed for Section XI, Revision 7

QC Engineering Transmittal Letter

Addition of a Nominal 3 Second Time Delay to EHC Low Pressure Turbine Trip Logic, September 9, 1998

System Planning Operating Guides

1-1	Generating Stations Operating Voltage Levels, Revision 2

1-1-A	Operating Nuclear Stations at Reduced Excitation Levels, Revision 2

10 CFR 50.59 Evaluations and Screenings

SE-97-151	Clarify updated final safety analysis report Description of the Residual Heat Removal Service Water pumps in Table 9.2-1, 11/12/1997
SE-98-088	Remove and Install New Type of Time Delay Relays in the Unit 1, Unit 2 and 1/2 Diesel Rooms
SE-98-098	Modify the Unit 1 Emergency Diesel Generator Fuel Transfer Pump Circuit to Allow Operation at Both Diesel Rooms
SE-98-099	Modify Turbine Trip Logic from Low Pressure Turbine Exhaust Hood High Temperature to Bypass the Trip Above 30% Turbine/Generator Power
SE-98-100	Modify Turbine Trip Logic (EHC Low Power) to Reduce the Probability of a Spurious Trip
SE-98-108	Modify Power Supply to the Scram Discharge Volume Level Switches
SE-98-110	Splice Station Blackout Control Cables and Reroute Power Cable Between 4 kV Switchgear 71 and 4 kV Switchgear 23-1
SE-98-151	Revise updated final safety analysis report Discussion on the Control Room Emergency Ventilation System,12/2/1998
SE-99-015	Revise Updated Final Safety Analysis Report on Minimum System Voltages and Location of Information, 2/26/1999
SE-99-016	Revise Updated Final Safety Analysis Report Section 5.3.2.2 to Allow ASME Section XI Pressure Testing of Non-Welded Components Using Nuclear Heat, 2/25/1999
SE-99-031	Correct Equipment Configuration Description Errors for the Station Blackout Diesel Generator System,6/3/1999
SE-99-056	Revise Updated Final Safety Analysis Report Table 6.2-7 to Reflect Correct Primary Containment Isolation Valves, 7/21/1999
SS-F-99-157	Enlarge Bore Sizes Of Reducing Orifices 2-3924 and 2-3925
SS-H-99-45	Revise Updated Final Safety Analysis Report Section 5.3.2.2 to Allow Pressure Testing at Power, 3/29/1999
SS-U-98-10	Correct Updated Final Safety Analysis Report Typographical Error on Product Name/Number, 2/11/1999
SS-U-98-11	Minor Wording Changes to Updated Final Safety Analysis Report Section 9.4.5.B, Page 9.4-8, 11/26/1998
SS-U-9902	Clarification of High Pressure Coolant Injection Steam Flow Switch Logic, 3/9/1999

Technical Specifications

3.8.C	Ultimate Heat Sink

Updated Final Safety Analysis Report Change Packages

97-R5-016	Clarify Updated Final Safety Analysis Report Description of the Residual Heat Removal Service Water Pumps in Table 9.2-1, 11/12/1997
97-R5-101	Minor Wording Changes to Updated Final Safety Analysis Report Section 9.4.5.B, Page 9.4-8, 11/26/1998
97-R5-103	Revise Updated Final Safety Analysis report to Show Technical Specification Allowed Range on Control Room Ventilation System Air Intake Value and Discuss Necessary Manual Operations, 12/2/1998
97-R5-115	Correct Vendor Product Name/Number, 2/13/1999
97-R5-118	Revise Updated Final Safety Analysis Report Section 5.3.2.2 to Allow Pressure Testing at Power, 3/29/1999

97-R5-119	Revise Updated Final Safety Analysis Report on Minimum System Voltages and Location of Information, 4/8/1999
97-R5-121	Clarification of High Pressure Coolant Injection Steam Flow, 3/11/1999
97-R5-130	Revise Updated Final Safety Analysis Report Sections 8.3.1.9.4.3, 8.3.1.9.4.4 & 8.3.1.9 to Correct Station Blackout Diesel Information, 6/7/1999
97-R6-005	Revise Updated Final Safety Analysis Report Table 6.2-7 on Containment Isolation Valves, 7/23/1999

Updated Final Safety Analysis Report Sections

Section 6.3.2.1.1	Core Spray Subsystem Interfaces with Other ECCS Subsystems
Section 6.3.2.1.2	Subsystem Characteristics
Section 6.3.2.2.1	LPCI Subsystem Interfaces with Other ECCS Subsystems
Section 6.3.2.2.2	Subsystem Characteristics
Section 6.3.2.3.1	HPCI Subsystem Interfaces with Other ECCS Subsystems
Section 6.3.2.3.2	Subsystem Characteristics
Section 6.4	Control Room Heating Ventilation and Air Conditioning
Section 7.2	Reactor Protection System
Section 7.2.3.1	Single Failure Criteria
Section 8.3.2	DC Power Systems
Section 8.3.3	AC Power Systems
Section 9.5.5	Diesel Generator Cooling Water System
Section 10.2	Turbine-Generator
Section 15.2.2.2	Load Rejection With Bypass
Section 15.2.3	Turbine Trip
Section 15.6.5.5.3	Control Room Dose Rates

Vendor Manuals

Barton	Model 227A Differential Pressure Indicator

White Papers

DG99-000135	Conduct of ASME Section XI Testing Following Replacements with Core Criticality at Boiling Water Reactors, 2/19/1999
254-100-96-01004	120VAC, 125VDC and 250VDC System Fuse/Breaker Coordination

Work Requests

980028462 01	Modify EHC Low Pressure Turbine Trip Logic ECTP #19
980044304 01	Perform Load Test for 24/48 Volt Battery
980075137 03	Perform Pre-fabrication Shop Work Required for Installation of New TD2 Relay
980083679 01	Low Pressure Turbine High Temperature Bypass Above 30% Power Per the Design Change Package
980084210 01	Manually Fill the Diesel Fire Pump Day Tank
980084652 06	Determinate/Terminate and Install Cables at Switchgear 71 in Station Blackout Building
980087963 02	Manually Fill the Diesel Fire Pump Day Tank
980087964 01	Modify Emergency Diesel Generator Fuel Oil Transfer Pump Logic
990005621 02	Revise Power Feed to Scram Discharge Volume Level Instruments