December 21, 2000

EA-00-262

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: QUAD CITIES NUCLEAR POWER STATION - NRC INSPECTION REPORT 50-254/00-18(DRS); 50-265/00-18(DRS)

Dear Mr. Kingsley:

On October 27, 2000, the NRC completed a baseline inspection at your Quad Cities Nuclear Power Station, Units 1 and 2. The preliminary results of that inspection were discussed with Mr. J. Dimmette, Jr. and members of your staff at the end of the inspection. Following the review of the preliminary findings by an NRC Significance Determination Panel (SDP), an additional discussion of our inspection findings was conducted with members of your Quad Cities Station and Corporate staff by telephone on November 27, 2000.

The enclosed report presents the results of this inspection. The inspection was an examination of activities conducted under your license as they relate to radiation protection and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, facility walk-downs and interviews with personnel. The inspection focused on as-low-as-is-reasonably-achievable (ALARA) planning and radiological work controls for the Unit 1 refueling outage (Q1R16).

This report discusses issues of low to moderate safety significance involving the failure to maintain radiation doses as-low-as-is-reasonably-achievable. As described in Section 2OS2.1 of this report, during Q1R16 the Safety Relief Valve (SRV) replacement job accrued more than 5 person-rem and exceeded the projected job dose by more than 50 percent as a result of a number of planning problems. This resulted in an apparent finding, which was assessed using the Occupational Radiation Safety Significance Determination Process (SDP), and was preliminarily determined to be White. White issues have some increased importance to safety and may require additional NRC inspection. This issue has a low to moderate safety significance because your 3-year rolling average, collective dose of 269 person-rem was greater than the 240 person-rem SDP screening criterion for the period 1997 through 1999, which is indicative of a continuing problem with radiation dose control. The dose for the SRV replacement job was originally estimated at approximately 18 person-rem and revised to 45 person-rem based on the as found radiological conditions. The final dose of 69.77 person-rem exceeded the re-estimated dose of 45 rem by more than 50 percent.

We have reviewed your post-job review which describes your evaluation of this job, and while we believe that we have sufficient information to make our final significance determination for this inspection finding, we are giving you the opportunity to provide us with additional information either in writing or at a regulatory conference. If you choose to provide additional information in writing, you should do so within 30 days of the date of this letter. Please contact Mr. Gary Shear at (630) 829-9876 as soon as possible, but within seven days of the date of this letter, to notify us of your intent. If we have not heard from you within the time specified, excepting a granted extension, we will assume that you agree with our evaluation of this matter.

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

Sincerely,

/RA/

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-254/00-18(DRS);
50-265/00-18(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
DCD - Licensing
J. Dimmette, Jr., Site Vice President
G. Barnes, Quad Cities Station Manager
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
W. Curtis, FEMA, Region V
E. Jenkins, FEMA, Region VII

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/RA/
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Division of Reactor Safety

Docket Nos. 50-254; 50-265
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E. Jenkins, FEMA, Region VII

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REGION III

Docket Nos:	50-254; 50-265
License Nos:	DPR-29; DPR-30
Report No:	50-254/00-18(DRS); 50-265/00-18(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Quad Cities Nuclear Power Station, Units 1 and 2
Location:	22710 206th Avenue North Cordova, IL 61242
Dates:	October 16-20, 23-27, and November 27, 2000
Inspectors:	J. E. House Senior Radiation Specialist M. W. Mitchell Radiation Specialist
Observer:	R. Alexander Radiation Specialist
Approved by:	Gary L. Shear, Chief Plant Support Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000254-00-18(DRS), IR 05000265-00-18(DRS), on 10/16-10/20, 10/23-10/27, and 11/27/2000, Commonwealth Edison Company, Quad Cities Nuclear Power Station, Units 1 and 2. ALARA planning and radiological controls for refueling outage Q1R16.

The inspection was conducted by a senior radiation specialist and a radiation specialist. The inspection identified one preliminary White finding. The significance of most/all findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “no color” or by the severity level of the applicable violation.

Cornerstone: Occupational Radiation Safety

- TBD. A variety of planning problems caused the dose for the Safety Relief Valve (SRV) replacement work completed during refueling outage Q1R16 to exceed its projected dose by more than 50 percent. During Refueling Outage Q1R16, significantly elevated dose rates were encountered in the drywell as a result of cobalt-60 plate out inside reactor coolant piping and steam lines. Work conditions were exacerbated because the drywell cooling/ventilation system was out of service for maintenance and testing, significantly elevating the temperatures in the drywell environment. While this environment was recognized, there was insufficient contingency planning for the lack of drywell cooling for the outage. In addition, less experienced workers performed the SRV job. As a result of these factors, radiation worker dose for the SRV job was not maintained as-low-as-is-reasonably-achievable (ALARA).

  The dose for the SRV replacement job was originally estimated at 18.85 person-rem. The final dose accrued was 69.77 rem which exceeded the re-estimated dose of 45 person-rem (based on the as-found radiological conditions) by more than 50 percent. Additionally, the licensee’s 3-year average collective dose (269 rem) was greater than 240 person-rem. Using the Occupational Radiation Safety Significance Determination Process, the NRC has made a preliminary determination that the finding was of low to moderate risk significance (White).

Report Details

Summary of Plant Status

During this inspection, Unit 1 was shut down and in a scheduled refueling outage. Unit 2 was at 95 percent power throughout the inspection period.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Controls for Radiologically Significant Areas

.1 Plant Walkdowns, Radiological Boundary Verifications and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors conducted walkdowns of the radiologically protected area (RPA) to verify the adequacy of radiological area boundaries and postings including high and locked high radiation areas in the Unit 1 and 2 Reactor Buildings, Turbine and Radwaste Buildings. Confirmatory radiation measurements were taken to verify that these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR 20, licensee procedures and Technical Specifications. Radiation work permits (RWPs) for higher dose jobs were reviewed for protective clothing requirements and electronic dosimetry alarm setpoints.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's condition report (CR) database and selected CRs related to radiological incidents, radiation worker performance, radiation protection technician performance, radiation work (radwork) practices and high radiation area access controls covering the previous six months. The inspectors evaluated the effectiveness of the radiation protection self-assessment process to identify problems and trends, and to implement corrective actions.

b. Findings

No findings of significance were identified.

.3 Job-In-Progress Reviews

a. Inspection Scope

The inspectors examined selected high dose jobs including the two highest dose jobs for the outage. RWP requirements and ALARA briefing packages were reviewed. Pre-job briefings and shift turn over meetings were attended. Dosimetry placement, job site radiological surveys, contamination controls, barricades and postings were reviewed. Enhanced job controls including electronic dosimetry and stay-times were evaluated.

b. Findings

One preliminary White finding for the Safety Relief Valve Replacement job, Radiation Work Permit (RWP No. 001012) is described in Section 2OS2.1

.4 High Risk Significant, High Dose Rate HRA and VHRA Controls

a. Inspection Scope

The inspectors reviewed the licensees controls for elevated dose rate areas and confirmed that locked high radiation areas were secured. There were no Performance Indicator occurrences for this area.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

.1 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors reviewed jobs being performed in areas of elevated dose rates and examined the work sites. Job exposure estimates were reviewed, work areas were surveyed to determine radiological conditions including low dose waiting areas and use of shielding. The ALARA briefing documentation was reviewed and the use of engineering controls was evaluated. During job site walk downs, radworkers and supervisors were observed to determine if low dose waiting areas were being used appropriately, and the effectiveness of job supervision including equipment staging, availability of tools and work crew size was evaluated.

b. Findings

One preliminary White finding was identified.

During Refueling Outage Q1R16 (October 13-31, 2000) significantly elevated dose rates were encountered in the drywell. Cobalt-60 plate-out on the inside of reactor coolant and main steam piping resulted in much higher outage dose rates than had been projected during the ALARA and work planning phase of outage preparation. After determining the actual drywell dose rates, the licensee recalculated the dose projections to account for the increased source term.

Removal and replacement of the Electromatic Relief Valves (ERV)/Safety Relief Valves (SRV)/Target Rock valves (RWP 001012, U1 ERV/SRV/Target Rock Valves: Remove Replace) resulted in a much higher dose than was anticipated during the ALARA/work planning process. This job consisted of removing and replacing relief valves attached to main steam lines. Work space at the job site, which is located on the second level of the drywell, was limited. The original dose estimate based on previous job performance was 18.85 person-rem. The final dose accrued for the job was 69.77 person-rem.

After shutdown, the licensee re-estimated the dose for this job (October 16) at 45 person-rem based on the as found elevated general area dose rates. As the job progressed (October 23), the licensee revised its re-estimate to 59.2 person-rem. This included an additional 3.9 person-rem due to the elevated drywell dose rates, 8.4 person-rem because worker stay times were reduced due to heat stress, which impacted worker efficiency, and 1.9 person-rem for contamination control which required additional RP technician coverage. On October 25, another re-estimate of 72.5 person-rem was made. This included an additional 2 person-rem for dose rates, 2.1 person-rem for heat stress, 3.8 person-rem for rework and 5.4 person-rem for worker "inefficiencies."

The licensee's dose projection of 45 person-rem, which was based on the elevated dose rates, is being used by the NRC as the ALARA baseline projection for the SRV replacement job, as no changes to work scope or radiation levels were identified subsequent to the 45 rem estimate. The October 16 re-estimate of 45 person-rem was based on elevated general area dose rates alone. The October 23 re-estimate of 59.2 person-rem included 8.4 person-rem for heat stress and 1.9 person-rem for contamination control. The heat stress should have been anticipated since the drywell cooling/ventilation system was planned to be out of service. Contingency planning should have addressed this issue. Contamination control should have been part of the ALARA planning process for breaching the steam system since it was known that moisture carry over had occurred and that the steam lines had significant unanticipated dose rates. There was no attempt to "stand down" and assess the full impact of the elevated dose rates, the potential causes and the radiological effects on workers. The October 25 re-estimate included additional estimates for heat stress and dose rates. It also contained 3.8 person-rem for rework and 5.4 person-rem for "inefficiencies." The additional dose due to rework could have been substantially reduced since it was known that the work crew performing the job was inexperienced. While inexperienced workers were paired with more experienced staff, more substantive training, including mock-ups, should have been used to reduce dose. The licensee attributed an additional 5.4 person-rem to "inefficiencies." However, these "inefficiencies" were not defined, and given the effect of heat stress and worker inexperience, could have been anticipated.

The dose for this job (69.77 person-rem) exceeded the projected dose (45 person-rem) by more than 50 percent and was greater than five person-rem. Additionally, The licensee's 3-year average collective dose (269 person-rem) was greater than the

240 person-rem threshold contained in the applicable “Issues Affecting Cornerstones” (Group 2 Questions), Inspection Manual Chapter (IMC) 0609. This information was further evaluated using the Occupational Radiation Safety (ALARA) Significance Determination Process (SDP) contained in IMC 0609 Appendix C, and preliminarily determined to be a White finding.

During the outage inspection, the inspectors observed the following ALARA/work planning weaknesses:

* There were insufficient provisions for drywell cooling or ventilation for the outage which resulted in elevated drywell temperatures. The cooling and ventilation systems were planned to be out of service during the early part of the outage for maintenance/testing; however, they were unavailable for most of the SRV project;

* Heat stress was a major issue which resulted in shortened stay times of approximately 35 minutes. This reduced worker efficiency;

* Work crews for the SRV job were relatively inexperienced compared to previous outages and mock-up use was limited; and

* Some ALARA pre-job briefings were conducted up to several weeks prior to the start of the outage.

Heat stress was a major contributor to the elevated dose incurred by workers. The drywell coolers were out of service for maintenance and testing during most of the outage; however, the licensee planned for the systems to be inoperable for approximately three days. The licensee’s work planning did not account for the lack of drywell cooling and ventilation, which resulted in the reduction of the SRV replacement work crew stay times to approximately 35 minutes per entry with the resultant drop in worker efficiency.

For the last several outages, mechanical maintenance personnel had performed the SRV job. During this outage, the station construction department personnel were utilized. The licensee estimated that approximately 30 percent of the craft work force did not have recent nuclear experience and lacked familiarity with the work location and job specifics. This lack of knowledge and experience resulted in workers spending more than the expected man-hours in high dose areas. Along with worker inexperience, mockup training did not adequately familiarize workers with plant equipment and layout, the use of tools and the techniques to reduce dose. This was evidenced by errors which resulted in rework. Examples of this were improper installation of nuts/bolts, installation of one SRV 180 degrees out of position, and old valve tagging left on newly installed valves.

The two major factors that resulted in reduced worker efficiency (and elevated dose to workers) were heat stress and worker inexperience. The licensee failed to fully consider these factors affecting worker efficiencies in the revised plans. Instead, the licensee increased its dose estimate as work progressed and previous estimates were exceeded. The inspectors noted an urgency to adhere to the outage schedule, which may have resulted in the licensee failing to fully consider worker efficiency under these environmental conditions and to implement measures to improve work performance (i.e., reduce worker dose).

.2 Radiation Worker Performance

a. Inspection Scope

The inspectors observed radiation worker (radworker) performance including the use of low dose waiting areas and proper use of protective clothing based on RWP requirements for higher dose jobs. Radiological conditions were discussed with radworkers to determine worker awareness of significant radiological conditions and electronic dosimetry set points. Radiological problem condition reports were reviewed to determine if weaknesses in radworker performance had been identified.

b. Findings

No findings of significance were identified.

.3 Radiation Protection Technician Performance

a. Inspection Scope

Radiation protection technician (RPT) performance was reviewed. This included job coverage, control of contamination and exit boundaries during job evolutions, control of radworkers and RPT response to radiological incidents. Radiological problem condition reports were reviewed to determine if RPT errors had been identified.

b. Findings

No findings of significance were identified.

.4 Radiological Work Planning

a. Inspection Scope

The inspectors reviewed the top five jobs based on radiological dose. Exposure estimates were compared with the accrued dose for each job to determine if any jobs had an actual dose that was 50 percent greater than the estimate, and if so did the job exceed 5 rem. Job planning (ALARA) was evaluated based on exposure estimates along with dose mitigation efforts which included time, distance and shielding. Coordination among operations, radiation protection, chemistry, maintenance, work planning/scheduling and engineering was evaluated along with person hour estimates to determine if dose mitigation was an integral part of the work planning/scheduling process. Job scheduling to take advantage of activities where time/distance/shielding could reduce dose was reviewed.

b. Findings

One NRC identified preliminary White finding is discussed in 2OS2.1.

.5 Verification of Exposure Estimate Goal and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the process for estimating annual radiological exposure. This included comparing actual exposure results with initial estimates, reviewing the exposure tracking system, and report timeliness and detail. Radiation work permits were reviewed to determine if job specific exposure trends could be identified. Management control of radiological work using radiation exposure was evaluated.

b. Findings

No findings of significance were identified.

.6 Declared Pregnant Workers

a. Inspection Scope

The inspectors reviewed the controls that would have been implemented by the licensee had a worker voluntarily declared a pregnancy in the last 11 months. Specifically, the inspectors reviewed the licensee's adherence with the requirements contained in 10 CFR 20.1208 by examining the licensee's fetal protection program procedures for tracking doses to the embryo/fetus, and the administrative and ALARA controls that could be used by the licensee to minimize the dose to the embryo/fetus of a declared pregnant worker.

b. Findings

No findings of significance were identified.

.7 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed audits, self-assessments and condition reports related to the ALARA program. Post outage reviews of higher dose jobs and critiques of the ALARA program were evaluated to determine if problems were identified and properly characterized, prioritized and entered into the corrective action program. The inspectors also reviewed Nuclear Oversight (NO) outage field observation reports, and outage generated condition reports to assess the adequacy of the licensee's ability to identify problems. Radiological work was examined for jobs that resulted in more than five person-rem and had a collective exposure of more than 50 percent over the exposure estimate. A post outage review and critique of the most dose intense job was evaluated to determine if radiological work problems/deficiencies had been identified, an adequate safety evaluation performed, and the problems entered into the licensee's corrective action system.

b. Findings

No findings of significance were identified for this area.

2OS3 Radiation Monitoring Instrumentation

.1 Respiratory Protection - Self Contained Breathing Apparatus (SCBA)

a. Inspection Scope

The inspectors reviewed the status and surveillance records for SCBA located in various areas onsite, with particular attention to those SCBA reserved for fire brigade and control room personnel. In addition, the inspectors verified that applicable emergency response and control room personnel were properly trained, mask fit, and medically qualified in the use of SCBA.

b. Findings

No findings of significance were identified for this area.

4. Other Activities

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Dimmette, Jr., and other members of licensee management and staff at the conclusion of the inspection on October 27, and with licensee and corporate management on November 27, 2000. The licensee acknowledged the information presented. During the November 27, 2000, telephone discussion, licensee representatives stated that they did not agree with the preliminary White finding identified by the NRC. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

E. Anderson, Radiation Protection Manager
D. Barker, Radiation Protection
G. Barnes, Station Manager
W. Beck, Executive Assistant
P. Behrens, Chemistry Manager
K. Bethard, Regulatory Assurance
R. Blaine, Radiation Protection, Corporate
G. Boerschig, Engineering Manager
R. Chrzanowski, Nuclear Oversight Manager
J. Dimmette, Jr., Site Vice President
T. Fuhs, Regulatory Assurance
D. Harmon, Systems Engineering
R. Hebeler, Chemistry Supervisor
D. Hieggelke, Nuclear Oversight Lead Assessor
D. Kallenbach, Radiation Protection
M. McDowell, Operations Manager
R. Norris, Radiation Protection Supervisor, Dresden Station
K. Ohr, Radiation Protection Supervisor
M. Perito, Maintenance Manager
C. Peterson, Regulatory Assurance Manager
G. Powell, Radiation Protection Supervisor
G. Rankin, Radiation Protection
J. Siper, Director of Licensing and Compliance
J. Sirovy, Nuclear Oversight Assessor
R. Svaleson, Shift Operations Superintendent
J. Woolridge, Radiation Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

One preliminary White finding for ALARA/work planning is discussed in section 2OS2.1

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably Achievable
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
ERV	Electromatic Relief Valve
HRA	High Radiation Area
IMC	Inspection Manual Chapter
NRC	Nuclear Regulatory Commission
OA	Other Activities
OS	Occupational Radiation Safety
PI	Performance Indicator
PIF	Problem Identification Form
Q1R16	Unit 1 Refueling Outage 16
Radwork	Radiation Work
RP	Radiation Protection
RPA	Radiologically Protected Area
RPT	Radiation Protection Technician
RWP	Radiation Work Permit
SDP	Significance Determination Process
SRV	Safety Relief Valve

PARTIAL LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection. 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

Condition Reports

Q2000-03642, Q2000-03768, Q2000-03644, Q2000-03515, Q2000-03519, Q2000-03633, Q2000-03634, Q2000-03715, Q2000-03718, Q2000-03719, Q2000-03831, Q2000-03877, Q2000-03821, Q2000-03775, Q2000-03757, Q2000-03932, Q2000-03933, Q2000-03514, Q2000-03886, Q2000-03788, Q2000-03879, Q2000-03716, Q2000-03766

Self-Assessments

Nuclear Oversight Field Observation Reports:
- 34687-18: Fuel Bundle Mis-Orientation
- 34687-07: Refuel Floor Activities

Focus Area Self-Assessment, AD-AA-103, Revision 3, “ALARA Planning and Controls”
Scorecard and Tour Data for September, 2000
Q1R16 Refuel Floor Airborne Contamination Event
Prompt Investigation for CR Q2000-03636, October 16, 2000, “Reactor Head Vent Disassembly Results in Airborne Radioactivity and Work Stoppage on Refuel Floor”
Prompt Investigation for CR Q2000-03821, October 22, 2000, “Multiple Contaminations Under Reactor Vessel When Installing LPRM Flush Cans”
Quad Cities Station ALARA Review, “Remove Replace Unit 1 ERV/SRV/Target Rock Valves”

Procedures

QCAP 0600-07 Unit 1(2), Revision 5, “Administration of the Radiation Protection Aspects of Quad Cities Fetal Protection and Post-Natal Programs”
QCRP 5510-21 Unit 1(2), Revision 10, April 8, 1999, “Maintenance and Inspection of the MSA Self-Contained Breathing Apparatus (SCBA)”

Radiation Work Permits

RWP 001012, Revision 1; U1 ERV/SRV Target Rock Valves: Remove/Replace
RWP 003514, Revision 2; U1 Flow Accelerated Corrosion: Replace/Repair Pipe
RWP 001042, Revision 2; U1 Drywell: Weld Overlays
RWP 001020, Revision 0; Control Rod Drives: Remove/Replace
RWP 001046, Revision 1; MSIP Weld Treatment
RWP 001031, Revision 1; LPRMs and SRM/IRM Dry Tubes: Under Vessel Work
RWP 003566, Revision 1; U1 Main Turbine: Overhaul/PM
RWP 001045, Revision 2; ISI Drywell Inspections
RWP 003581, Revision 0; U1 Reactor Disassembly/Reassembly, Cavity Work/Wall Cleaning

Miscellaneous

Unit 1 Drywell Survey Maps
Reactor Water Radio-Nuclide Analyses
Unit 1 Reactor Nuclide Trending
Respiratory Qualification Charts
2000 Source Term Reduction Plan, Quad Cities Station
Use of Cobalt Free Alloys in Valves, Revision 0, July 7, 2000
List of Completed/Planned Hydrolase Jobs
List of Hot Spots: Relative Dose Effect