U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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| Docket No: | 50-302 |
| License No: | DPR-72 |
| Report No: | 50-302/98-08 |
| Licensee: | Florida Power Corporation |
| Facility: | Crystal River 3 Nuclear Station |
| Location: | 15760 West Power Line Street<br>Crystal River, FL 34428-6708 |
| Dates: | August 2 through September 12, 1998 |
| Inspectors: | S. Cahill, Senior Resident Inspector<br>S. Sanchez, Resident Inspector<br>S. Ninh, Project Engineer (Sections 08.1-08.4) |
| Approved by: | L. Wert, Chief, Projects Branch 3<br>Division of Reactor Projects |



Enclosure

EXECUTIVE SUMMARY

Crystal River 3 Nuclear Station
NRC Inspection Report 50-302/98-08

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a six-week period of resident inspection; in addition, it includes the results of open item reviews by a regional project engineer.

Operations

- Hurricane preparations were thorough and proactive. Although tropical storm levels were not exceeded, the licensee was prepared for an upgrade to a hurricane watch and exhibited good coordination between departments to restore an out-of-service emergency diesel (Section O1.1).

- An unplanned manual reactor trip occurred due to a personnel error in control of troubleshooting, causing a main steam isolation of the A steam generator. Very few equipment problems occurred on the trip and the licensee effectively completed forced outage work and planned power reduction work during the shutdown (Section O1.2).

- The licensee's plant startup activities were deliberate and well-controlled. The licensee was self-critical and thorough in dispositioning issues to ensure incorporation into future forced outage schedules (Section O1.3).

- A revision to a surveillance test to address a previous problem was not completed prior to the next scheduled performance. The previous problem was not discussed in the pre-job brief. A formal operability resolution had to be reactively generated when the problem recurred (Section O2.1).

- The development of a routine Quality Assurance audit review item on the Post-Accident Sampling System into an aggregated report on several problems with the system was indicative of a thorough and comprehensive inspection (Section O7.1).

Maintenance

- Technicians performing a reactor protection system surveillance were knowledgeable of past problems and sensitive to avoiding similar circumstances. This was considered indicative of effective corrective actions (Section M1.1).

- Diesel generator 1B maintenance and modification work was precisely controlled. Pre-job planning, scheduling, and coordination were highly detailed and were aided by use of the system engineer as a single point-of-contact. Work suspension and diesel recovery to available status due to adverse weather warnings were significantly enhanced by this preplanning. Excellent sensitivity was displayed toward minimizing diesel out-of-service time (Section M1.2).

- An at-power reactor building entry was closely coordinated, supervised, and planned. The lack of a specific foreign material exclusion checklist contributed to a radio being inadvertently left in the RB and another entry was necessary to retrieve it (Section M1.3).

Engineering

- Engineering processes were upgraded to correct previous deficiencies. The licensee completed transition of the Configuration Document Integration Project into the normal design process, placed licensing basis archive information on a computer network, and implemented an engineering calculation cross-reference system. These enhanced engineering's ability to perform design reviews and safety assessments and maintain accurate design basis documents (Section E1.1).

Plant Support

- A quarterly licensee emergency plan drill was observed. Licensee drills continue to be diverse and effective training exercises (Section P1.1).

- Health Physics staff effectively controlled doses for an at power reactor building entry. Expected conditions the workers would encounter were understood in detail (Section R1.1).

- Use of drinking fluids for high heat stress in a Radiation Control Area (RCA) for the reactor building entry was administered well, but the lack of specific documented controls for their use and an exception to the RCA prohibition on drinking had not been questioned by licensee Health Physics personnel supervising the entry (Section R1.1).

- Inconsistent expectations were promulgated to plant personnel on the treatment of contaminated area boundaries (CAB) when a fire watch was observed reaching across a CAB to electronically read a bar code. Actions by Health Physics to move the CAB and relocate the bar code were considered appropriate (Section R1.2).

## Report Details

Summary of Plant Status

The plant began the inspection period at full rated thermal power and remained at that level until the reactor was manually tripped on August 27, 1998, following an inadvertent closure of the A steam generator main steam isolation valves due to a troubleshooting error (Section O1.2). The reactor was restarted and tied to the electrical grid on August 29, 1998. The plant returned to full power on August 30, 1998, and operated at that level for the remainder of the report period.

## I. Operations

### O1 Conduct of Operations

#### O1.1 General Comments (71707)

Using Inspection Procedure 71707 the inspectors performed routine reviews of plant operations which included shift turnovers, response to emergent problems, operator log reviews, coordination meetings, extended observations of operator conduct, and system walkdowns. Significant observations are discussed in the following paragraphs.

In May of 1998, the Institute for Nuclear Power Operations (INPO) conducted the periodic onsite evaluation of the licensee. The inspector reviewed the INPO report for that evaluation which delineated the final assessment findings. The inspector noted that the report findings were consistent with previous Nuclear Regulatory Commission (NRC) assessments. No new safety issues were noted.

On September 2, 1998, the licensee implemented their Emergency Response Procedures due to the declaration of a Tropical Storm Warning from Hurricane Earl. The inspector observed the licensee's preparations and response to this situation and attended the Adverse Weather Committee meetings. The licensee closely monitored storm location and conditions and looked ahead to needed actions if the condition were upgraded to a hurricane warning. At the time the warning was received, the B emergency diesel generator (EDG) was out of service for several maintenance activities and a modification to the cooling system vents. The inspector observed that the licensee quickly backed out of the work. Good coordination was exhibited between Operations, Maintenance, and Engineering departments during restoration of the EDG alignment and completion of post-modification and maintenance testing. The licensee recovered the EDG to an available status but did not run the 4-hour loaded surveillance test to declare it procedurally operable due to the risk associated with running the test and connecting the EDG to the electrical transmission grid during the unstable tropical storm conditions. The inspector considered these actions appropriate and verified the licensee took timely actions to restore, secure, and protect other vital equipment. The storm did not cause any plant complications and the inspector concluded that the licensee's preparations were thorough and proactive.

O1.2 Manual Reactor Trip due to Inadvertent Main Steam Isolation Valve Closure

a. Inspection Scope (93702, 71707)

The inspectors responded to the site and verified the licensee's actions in response to a manual reactor trip on August 27, 1998, at 11:51 p.m. The inspectors also reviewed preliminary licensee investigation results.

b. Observations and Findings

The inspectors reviewed the operational parameter data collected by the licensee and verified a manual reactor trip had been promptly initiated following closure of the A once-through steam generator (OTSG) steam isolation valves. The plant responded as expected and the inspectors did not identify any operational concerns. The inspectors observed that response of the plant equipment from the transient was understood in detail by operators.

The licensee identified an error in troubleshooting of the emergency feedwater initiation and control (EFIC) circuitry as the cause of the trip. The inspectors interviewed operators and engineers, reviewed the formal troubleshooting plan, and reviewed EFIC electrical and logic schematics to verify the licensee's conclusion. The error in the troubleshooting plan involved an omitted action to disable the actuation capability of the EFIC channel being tested. Although the licensee has a formal troubleshooting procedure process and this plan had been developed and approved by that process, the omitted action step was not recognized by the maintenance supervisor developing the plan or the engineer and Nuclear Shift Manager approving it. Several other factors were contributing causes to the troubleshooting plan omission and verification reviews, but the inspector noted these were already identified and being prioritized appropriately by the licensee. The licensee was processing Licensee Event Report (LER) 50-302/98-09-00 to describe their findings and finalized corrective actions.

The inspectors observed that the licensee effectively integrated work for a down power evolution that had been planned for August 29, 1998 into the trip outage window schedule. The licensee also integrated mandatory items from their forced outage work lists into the schedule. The inspector reviewed items that were selected and those that were deferred to a later outage and did not identify any problems with the licensee's prioritizations. One noteworthy forced outage item was a reactor building entry which corrected a recently identified reactor coolant pump (RCP) 1A oil leak.

c. Conclusions

An unplanned manual reactor trip occurred due to a personnel error in control of troubleshooting, causing a main steam isolation of the A steam generator. Very few equipment problems occurred on the trip and the licensee effectively completed forced outage work and planned power reduction work during the shutdown.

O1.3 Plant Recovery From Reactor Trip (71707)

The inspector reviewed the licensee's post reactor trip evaluation prior to startup. The inspector also attended the Plant Review Committee (PRC) meeting that discussed the results of the post trip evaluation. The PRC determined the following needed to be dispositioned before startup: replace the failed EFIC component; develop temporary instructions to require a second party review prior to the use of the troubleshooting procedure; and, review the lessons learned with Maintenance personnel. The inspectors considered these conclusions appropriate.

The inspector observed the licensee's startup activities and determined that they were performed in a deliberate and well-controlled manner. A problem was noted with the estimated critical position (ECP) calculations. During the required review by the senior reactor operator (SRO), discrepancies were found in the desired rod bands between the two ECPs provided by reactor engineering. The ECP was corrected prior to reactor startup and the inspectors considered that the ECP review process barriers in place worked to identify the problem. The licensee recognized room for improvement in the ECP verification process so that an incorrect ECP would not be handed to an SRO without adequate verification of its validity. This issue was actively being pursued by the licensee. No other concerns or problems were noted during startup.

The inspector also attended the initial forced outage critique. The most notable item discussed was the duration of work performed during the forced outage. Each job that exceeded or fell short of its scheduled duration was discussed to determine if the times needed to be adjusted for future outage schedules. The licensee was self-critical and thorough in its disposition of issues that arose to ensure incorporation into future forced outage schedules.

## O2 Operational Status of Facilities and Equipment

### O2.1 Emergency Feedwater Valve EFV-35 Failed Acceptance Criteria

a. Inspection Scope (71707)

On August 11, 1998, during the performance of an Emergency Feedwater Pump (EFP) 2 and valve surveillance procedure (SP), the inspector observed emergency feedwater valve (EFV)-35, the motor driven emergency feedwater pump (EFP-1) discharge check valve, failed to meet acceptance criteria. This same failure previously occurred in May 1998 as discussed in Inspection Report 50-302/98-06. The inspector reviewed the licensee's response to the earlier problem.

b. Observations and Findings

The surveillance required valve EFV-35 be tested to verify sufficient closure to limit backflow to pump EFP-1 while EFP-2 was running. Acceptance criteria for EFV-35 closure was less than or equal to 200 pounds per square inch gauge (psig) upstream of the valve. The actual pressure recorded was 530 psig. EFV-35 was declared inoperable, while pump EFP-2 was determined to be operable based on discussions with Engineering. These discussions determined that the required developed head for EFP-2 would be available despite the leakage past EFV-35. An Operability Concerns Resolution (OCR) was reactively generated to formally document the operability determination basis. This OCR was reviewed by the inspector with no concerns noted.

However, the inspector determined that a corrective action for the previous EFV-35 failure, to revise the SP to remove the acceptance criteria based on an engineering evaluation, was not completed before this scheduled SP performance. This evaluation showed that a complete failure of EFV-35 would not hinder EFP-2 from delivering the required flow to the OTSG. The inspector also noted that the pre-job brief did not discuss this previous failure, even though the same crew conducted both this and the previous SP. Although the procedural revision to remove EFV-35 from the acceptance criteria was waiting on the formal issuance of the engineering calculation, Operations management indicated that their revision of the SP should have been more prompt and that the pre-job brief could have averted the need to reactively generate an OCR had the previous operating experience been discussed. Regulatory and procedural requirements were met.

c. Conclusions

A revision to a surveillance test to address a previous problem was not completed prior to the next scheduled performance. The previous problem was not discussed in the pre-job brief. A formal operability resolution had to be reactively generated when the problem recurred.

## O7 Quality Assurance in Operations

### O7.1 Licensee Quality Assessment Group Activities (71707, 40500)

The inspectors routinely reviewed the activities and results of the licensee's Nuclear Quality Assessments (NQA) group. Special Assessment Report QPS-98-0068 on the Post-Accident Sampling System (PASS) was reviewed in detail. The report was initiated after a routine audit item developed into a larger issue. The inspectors noted that the report thoroughly assessed past and present problems with the system, effectively aggregated the problems to develop a broader conclusion on the condition of PASS, and researched numerous requirement and guidance documents. The NQA effort was effective in redirecting Engineering attention to the overall system condition vice correcting each problem individually. The report identified several problems and numerous areas for improvement for items such as incomplete procedures. The inspector verified that these were incorporated into the licensee's corrective action program (CAP) as required.

The inspectors also reviewed the results of the licensee's self-assessment of their NQA program performed by NQA counterparts from other licensees. The inspector found the assessment results consistent with previous NRC observations that NQA were effective auditors. However, one finding involved the role of the NQA department in the CAP, in that NQA was not ensuring that responses and corrective action plans to precursor cards (PC) identified by NQA were evaluated for adequacy. Although provisions existed in the licensee CAP system for these reviews, they had been suspended by NQA due to workload concerns. At the end of report period, the licensee was evaluating their expectations for NQA involvement in CAP.

## O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) VIO 50-302/98-01-06: Lack of Emergency Lights for Operation of Appendix R Safe Shutdown Equipment. The inspector reviewed the violation dated March 4, 1998, and the licensee's response in a letter dated March 31, 1998. The inspector verified that Emergency Lights were installed in the Control Rod Drive room. The inspector concluded that the licensee's corrective actions were appropriate and implemented. This violation was closed.

O8.2 (Closed) VIO 50-302/98-04-03: Failure to Follow Procedure CP-111 for Documenting, Evaluating, and Correcting Adverse Conditions. The inspector reviewed the violation dated June 8, 1998, and the licensee's response in a letter dated June 23, 1998. The inspector determined that plant personnel involved with review of the PC documenting discrepancies were counseled on the importance of processing manual PCs. Compliance Procedure CP-111. Processing of Precursor Cards for Corrective Action Program, was revised on April 18, 1998, to implement a new version of the Corrective Action Program database. This new version allows for electronic PC initiation (a unique number assigned to the PC), supervisor review, and Shift Technical Advisor approval. This reduces the probability of misplacing a PC in the review and approval process. The inspector verified that the licensee's corrective actions were appropriately implemented. This violation was closed.

O8.3 (Closed) IFI 50-302/98-02-03: EOP Enhancements. This Inspector Follow up Item (IFI) was opened to review the licensee's evaluation of Emergency Operating Procedure (EOP)-08, Loss of Coolant Accident (LOCA) Cooldown, step 3.97 through 3.100, and the licensee's additional training on Reactor Coolant System cooldown rates with a steam leak and Steam Generator Tube Rupture (SGTR) in the same OTSG. The inspector determined that the sequencing of step 3.97 through 3.100 of EOP-08 was evaluated and revised on April 8, 1998. The inspector verified that the additional training requested by the EOP group regarding simultaneous OTSG tube and steam leaks was completed on August 18, 1998 under the special training provisions of lesson plan ROT-05-101. Therefore, this IFI is closed.

O8.4 (Closed) LER 50-302/98-04-00: Unanalyzed Cross-Connecting of 480 Volt Busses. On February 12, 1998, the licensee identified 480 volt (V) electrical bus alignments that were not analyzed for the voltage drop or short circuit capability. The licensee determined that these electrical alignments to supply power to the non-safety related 480 V busses from the Engineered Safeguards (ES) busses might have been infrequently used during previous plant operation in Modes 1 through 4. The apparent cause for the electrical alignments without analysis was inadequate design control. As a result of the evaluation, the licensee decided to utilize existing calculations to set loading limits and permit use of 480 V tie breakers during Modes 5 and 6, and to procedurally prohibit the use of 480 V tie breakers during Modes 1 through 4. However, administrative controls will remain in place to prohibit supplying power to non-safety related 480 V busses from ES busses, and cross-connecting the 480 V ES busses until applicable plant procedures are revised to incorporate the above restrictions. Nuclear Engineering Procedure NEP-213, Design Analyses/Calculations, was revised to include a checklist for considering atypical equipment alignments when performing engineering calculations. The inspector determined that the licensee identified this issue while investigating the extent of condition for LER 302/98-02-00, "Use of 500 kilovolt Electrical Backfeed while not a Qualified Source of Off-site Power." The inspector considered that this issue constitutes an additional example of Violation 50-302/98-01-07 and is not being cited individually. No additional response to violation

50-302/98-01-07 is required. The inspector verified that corrective actions for this additional example had been taken in conjunction with corrective actions for the previously cited violation. Therefore, this LER is closed.

## II. Maintenance

### M1 Conduct of Maintenance

#### M1.1 General Comments

a. Inspection Scope (62707, 61726)

The inspectors observed all or portions of the following work requests (WR), surveillance procedures (SP), maintenance procedures (MP), preventive maintenance procedures (PM), and reviewed associated documentation:

- SP-110B "B" Reactor Protection System Functional Testing
- SP-130 Engineered Safeguards Monthly Functional Test
- SP-335A Radiation Monitoring Instrumentation Functional Test
- SP-146A Emergency Feedwater Initiation and Control Monthly Functional Test
- WR 0353945 [Waste Disposal System] WDRS-2 Rupture Disc Replacement
- MP-122 Disassembly and Reassembly of Flanged Connections
- PM-275 General Preventive Maintenance Work
- PM-178A Preventive Maintenance of Limitorque Actuators 2 Year Inspection

b. Observations and Findings

The inspector observed the performance of surveillance SP-110B on August 3, 1998, due to a prior concern with attention to detail when technicians performed this surveillance. In May of 1998, the licensee had identified that the reactor protection system (RPS) low pressure trip bistables were not adequately tested, and documented the problem in LER 50-302/98-06. The inspector had concluded that attention to detail during the testing was not strong because the problem was not identified earlier. During the August 3, 1998 performance of SP-110B, the inspector questioned the technicians on their knowledge of past problems with the performance of this series of SPs. The technicians were knowledgeable of past problems and sensitive to avoiding similar circumstances. This was considered indicative of effective corrective actions.

During the performance of surveillance SP-335A on August 22, 1998, the inspector noted a few minor procedure problems that in some cases were already identified as a NUPOST (Nuclear Procedures Observations/Suggestions Tracking System) comment that were awaiting a revision to the procedure. In the other cases, the operators properly addressed and corrected the problems at the time of identification.

The inspector observed maintenance work to replace the rupture disc on waste disposal system component WDRS-2, a containment penetration expansion chamber. The mechanics performing the work were generally knowledgeable of the task, but had not worked on this specific component before. The work occurred within a contaminated area and at one point was delayed while the assisting mechanic tried to locate a specific tool to complete the task. After about 40 minutes, the assisting mechanic returned without locating the tool. The mechanics determined that the tool would have to be made, so the work was temporarily stopped. The work resumed approximately one hour later, and the necessary tool was found at a different location than had previously been searched. The work went to completion with no other problems noted. After discussions with Maintenance supervision, the inspector determined that although applicable procedures had been followed, two processes to prevent poor job performance had not been completed properly. The work package preparation by the planner and the pre-job walkdown by the mechanics did not identify all the necessary tools.

c. Conclusions

Technicians performing a reactor protection system surveillance were knowledgeable of past problems and sensitive to avoiding similar circumstances. This was considered indicative of effective corrective actions.

M1.2 Emergency Diesel Generator (EGD) 1B Maintenance and Modification

a. Inspection Scope (62707, 71707)

On September 2, 1998, the licensee removed the 1B EDG from service for routine testing, corrective maintenance, and implementation of a modification to the jacket cooling system vents. The inspector reviewed the scope and scheduling of the work, attended coordination and pre-job briefings, verified equipment clearances were correctly implemented, and observed work in progress.

b. Observations and Findings

The inspector attended a pre-job coordination meeting the day before the work started and observed that the licensee had created an integrated schedule for all of the EDG activities that controlled the time in the EDG Technical Specification (TS) Limiting Condition for Operation (LCO) down to hour increments. The EDG system engineer had been assigned as a single point of contact to coordinate the simultaneous performance of several corrective maintenance items, painting of EDG components, surveillance testing, and the vent line modification. The inspector observed that all involved groups attended this meeting, the schedule was discussed in detail and agreed upon by all parties, individual responsibilities were clear, and that an overriding goal was to limit the time in the LCO. Most of the modification work had been completed in the shop and pre-staged to minimize the EDG outage duration.

On September 2, 1998, the inspector verified that Tagging Clearance Orders 98-08-185 and 98-08-017 were properly implemented to secure the EDG for the work and that the entry time into the LCO coincided with tagging out the EDG. The inspector did not identify any discrepancies with these items. The inspector observed the work in progress and noted it correlated with the pre-arranged schedule, was done with work packages present and in active use, and was worked diligently. The work had to be suspended due to adverse weather conditions as discussed in Section O1.1, but the inspector observed that the detailed pre-planning and coordination allowed the EDG to be restored to an available status within a few hours.

c. Conclusions

Diesel generator 1B maintenance and modification work was precisely controlled. Pre-job planning, scheduling, and coordination were highly detailed and were aided by use of the system engineer as a single point-of-contact. Work suspension and diesel recovery to available status due to adverse weather warnings was significantly enhanced by this preplanning. Excellent sensitivity was displayed toward minimizing diesel out-of-service time in a Technical Specification LCO.

M1.3 Reactor Building Entry at Power to Add Oil to RCP-1A

a. Inspection Scope (62707)

On August 27, 1998, the licensee sent individuals into the reactor building (RB) to add oil to RCP-1A to makeup for a leak and to verify the status of a decay heat system (DH) valve leak. The inspector observed the control of the entry and the performance of airlock testing and containment inspection surveillance procedures. The inspector also attended a post-job critique.

b. Observations and Findings

The inspector observed that the RB entry and oil addition activities were closely coordinated with control room operators, were constantly supervised by a Maintenance supervisor, and were planned in detail. During observation of SP-324, Containment Inspection, the inspector noted that the extent of foreign material exclusion (FME) controls for the RB was a single comprehensive sign off that all items brought in were removed. The inspector observed the Maintenance Supervisor performing the SP paperwork query the individuals collectively as they left the RB if they had removed all items so he could sign the SP step. A short while later, one of the individuals who had been in the RB returned to the access area looking for a radio he suspected he had left in the RB. Operators who were performing SP-430, Containment Airlock Seal Leakage Test, reentered the RB and retrieved the radio just inside the airlock. The inspector considered that the lack of a specific itemized FME checklist for items brought into the area contributed to this oversight. The inspector did not observe any problems with the performance of SP-430.

The forgotten radio was discussed at the post-job critique and the licensee was evaluating adding an FME checklist to SP-324. The post-job critique was thorough and included all concerns noted by the inspector. Inspector observations of Health Physics staff controls of the RB entry are discussed in Section R1 of this report.

c. Conclusions

The inspector concluded that an at-power reactor building entry was closely coordinated with control room operators, was constantly supervised, and was planned in detail. The lack of a specific foreign material exclusion checklist for the entry contributed to a radio being inadvertently left in the RB and another entry was necessary to retrieve it.

## III. Engineering

### E1 Conduct of Engineering

### E1.1 Design Control Processes and Tools (37551)

The inspectors reviewed several recent enhancements and upgrades to licensee Engineering Processes, including transition of the Configuration Document Integration Project (CDIP) into the normal design change process. The CDIP was the dedicated project implemented during the licensee design outage from September 1996 until February 1998 to ensure the numerous design changes were consistently incorporated in design basis documents and the Final Safety Analysis Report (FSAR). The licensee also instituted publishing of licensing correspondence and license basis archive information on their computer network and implemented a searchable computerized engineering calculation cross-reference system. The inspectors did not verify the adequacy of any one of these improvements in detail. However, in the aggregate they represented clear improvements in the tools available for engineers to perform effective design reviews and safety assessments and to maintain accurate design basis documents. These areas had been contributors to the cause of the licensee design outage. The inspectors concluded the licensee was continuing to appropriately improve their design control processes in response to outage lessons learned.

## IV. Plant Support

### R1 Radiological Protection and Chemistry (RP&C) Controls

#### R1.1 Health Physics Control of Reactor Building Entry

a. Inspection Scope (71750)

The inspector observed Health Physics (HP) control of an RB entry on August 27, 1998, as discussed previously in Section M1.3.

b. Observations and Findings

The inspector observed that licensee HP personnel exhibited good control of individuals' radiation doses. The inspector interviewed the attending HP personnel and determined the entries were planned in detail and dose levels at transit and work locations were well understood. This planning contributed to only a few millirem being received for the entire evolution. The inspector also observed that canned beverages were staged for the use of personnel exiting the RB. The inspector considered the use of fluids appropriate for this evolution due to the high heat stress level of the RB when at power, but questioned the method for controlling it. The controls observed by the inspector were adequate in that a dedicated maintenance worker monitored the beverages and would only dispense them once an individual had removed anti-contamination clothing and frisked in a personnel contamination monitor. However, when questioned by the inspector, the worker was unsure of the source of these controls and whether they were addressed on the Radiological Work Permit (RWP). The inspector questioned the HP technicians who were also unsure of the allowance for the beverages and indicated the maintenance worker was controlling it.

The inspector determined that the RWP for the RB entry did not delineate controls for the beverages and did not specifically address the radiologically controlled area (RCA) restriction prohibited eating or drinking in an RCA. Licensee HP management subsequently determined that their basic radiological safety procedure mentioned that fluids would be used in high heat stress conditions in an RCA but it said their use will be under "strict controls." These strict controls were specifically not delineated in any other HP or maintenance procedure. At the close of the report, the licensee was developing their expectations for these controls. Regulatory requirements regarding control of contamination were met during this evolution. However, the lack of specified controls and an exception to RCA requirements prohibiting drinking had not been questioned by the HP technicians supervising the entry.

c. Conclusions

Health Physics staff effectively controlled doses for an at power reactor building entry. Expected conditions the workers would encounter were understood in detail.

Use of drinking fluids for high heat stress in a Radiation Control Area (RCA) for the reactor building entry was administered well, but the lack of specific documented controls for their use and an exception to the RCA prohibition on drinking had not been questioned by licensee Health Physics personnel supervising the entry.

R1.2 Treatment of Contaminated Area Boundaries (71750)

The inspector observed fire watch patrols reach across Auxiliary Building contaminated area boundaries (CAB) to electronically read a tour routine bar code. This was contrary to previously promulgated HP expectations that a CAB plane was not to be breached. However, the inspector determined that a prior understanding existed between Security (fire watch patrols) and HP to allow this practice. No procedural requirements were violated and the probability of contamination problems was remote. A similar issue had previously been identified by the inspector when an operator was observed reaching across a CAB to shine a flashlight on an oil sight glass. At that time, HP promulgated the aforementioned expectations to plant personnel. Because of this recent issue, the inspectors were concerned that an inconsistent standard was being promulgated to plant personnel and that fire watch patrols did not fully understand the sensitivity of CABs. HP determined that a better practice would be to move the CAB or relocate the bar code rather than send inconsistent expectations. The inspectors considered these actions appropriate.

P1 Conduct of EP Activities

P1.1 Quarterly Emergency Plan Drill (71750)

The inspectors participated in the licensee's quarterly emergency plan (EP) drill conducted on September 2, 1998, in the Training Center

Simulator and the Technical Support Center (TSC). The inspectors observed that the drill scenario was unique and primarily focused on exercising Health Physics personnel. Licensee EP staff emphasis was on the training value of the drill as the inspectors had observed in previous quarterly drills. The inspectors did not observe any significant problems in the simulator or TSC, but did observe that drill announcements on the licensee public address system could not be heard in diesel generator radiator rooms. This problem and others identified by the licensee were being appropriately addressed by the licensee via their corrective action system. The inspector concluded the licensee's quarterly emergency plan drills are diverse and continue to be effective training exercises.

## V. Management Meetings

### X1 Exit Meeting Summary

The inspection scope and findings were summarized on September 14, 1998. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

## PARTIAL LIST OF PERSONS CONTACTED

Licensees

S. Bernhoft, Manager, Nuclear Licensing
J. Cowan, Vice President, Nuclear Operations
R. Davis, Assistant Plant Director, Operations and Chemistry
R. Grazio, Director, Nuclear Regulatory Affairs
G. Halnon, Director, Nuclear Quality Programs
B. Hickle, Acting Director, Nuclear Operations Training
J. Holden, Director, Site Nuclear Operations
M. Marano, Director, Nuclear Site & Business Support
C. Pardee, Director, Nuclear Plant Operations
W. Pike, Manager, Nuclear Regulatory Compliance
M. Schiavoni, Assistant Plant Director, Maintenance
J. Terry, Acting Director, Nuclear Engineering & Projects

NRC

S. Ninh, Project Engineer, Region II (September 8-11, 1998)
L. Wiens, Project Manager, NRR (September 1-3, 1998)

## INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving and Preventing Problems

IP 61726: Surveillance Observations
IP 62707: Conduct of Maintenance
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 92901: Follow up - Operations
IP 93702: Prompt Onsite Response to Events at Operating Power Reactors

## ITEMS OPENED, CLOSED, AND DISCUSSED

### Opened

| Type | Item Number | Status | Description and Reference |
|---|---|---|---|
| None | | | |

### Closed

| Type | Item Number | Status | Description and Reference |
|---|---|---|---|
| VIO | 50-302/98-01-06 | Closed | Lack of Emergency Lights for Operation of Appendix R Safe Shutdown Equipment. (Section 08.1) |
| VIO | 50-302/98-04-03 | Closed | Failure to Follow Procedure CP-111 for Documenting, Evaluating and Correcting Adverse Conditions. (Section 08.2) |
| IFI | 50-302/98-02-03 | Closed | EOP Enhancements. (Section 08.3) |
| LER | 50-302/98-04-00 | Closed | Unanalyzed Cross-Connecting of 480 Volt Busses. (Section 08.4) |

### Discussed

| Type | Item Number | Status | Description and Reference |
|---|---|---|---|
| LER | 50-302/98-09-00 | Open | Personnel Error During Troubleshooting Causes a Main Steam Line Isolation and Manual Reactor Trip. (Section 01.2) |
| LER | 50-302/98-02-00 | Closed | Use of 500kv Electrical Backfeed While Not a Qualified Source of Off-Site Power. (Section 08.4) |