UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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August 24, 2001

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER 3 - NRC SUPPLEMENTAL INSPECTION REPORT 50-302/01-07

Dear Mr. Young:

On August 8, 2001, the NRC completed an inspection at your Crystal River Unit 3 facility. The enclosed report documents the inspection findings which were discussed on August 9, 2001, with you and members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspector reviewed selected procedures and records, observed activities, and interviewed personnel. Specifically, this inspection reviewed activities related to increased reactor coolant system identified leakage. The leakage had been reported as a White performance indicator for May 2001, in the NRC's Reactor Oversight Process. The performance indicator was in the barrier integrity cornerstone of the reactor safety strategic performance area.

We found your actions in response to the increased reactor coolant system leakage to be appropriate. Corrective actions to address the leakage were completed in a timely manner. No findings of significance were identified during the inspection.

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 D. Monninger, Acting Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 50-302
License No. DPR-72

Enclosure: NRC Supplemental Inspection Report
50-302/01-07

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.	50-302
License No.	DPR-72
Report No:	50-302/01-07
Licensee:	Florida Power Corporation (FPC)
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Road Crystal River, FL 34428-6708
Dates:	August 6 - 8, 2001
Inspector:	Scott Stewart, Senior Resident Inspector
Approved by:	John D. Monninger, Acting Chief Reactor Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000302-01-07, on 08/06-08/2001, Florida Power Corporation, Crystal River Unit 3. Supplemental inspection for increased reactor coolant system leakage which was reported as a White performance indicator.

The inspection was conducted by the senior resident inspector. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Cornerstone: Barrier Integrity

This supplemental inspection was performed by the NRC to assess Florida Power Corporation's activities associated with increased reactor coolant system leakage. The leakage exceeded the NRC's Reactor Oversight Process White performance indicator threshold in May, 2001. The White indicator threshold is set at 50 percent of the leakage allowed by Technical Specifications and corresponds to a performance level that may result in increased NRC oversight.

Using Inspection Procedure 95001, the inspector determined that the licensee's root cause evaluation for the reactor coolant leakage was acceptable and that the licensee had taken appropriate corrective actions. The source of the leakage was seal ring leakage from a valve located between the reactor and the decay heat removal system. The licensee stated that the most likely cause of the valve leakage, initially identified in April 2000, was uneven seating forces around the pressure retaining valve seal ring. The uneven forces had likely resulted from either a 1999 valve rebuild or reactor coolant system pressure cycles associated with plant shutdown and subsequent heatup. A plant shutdown and restart in May 2001 was associated with a distinct increase in the leakage that caused the performance indicator to cross the White threshold. The leakage exceeded the White performance indicator threshold for about 24 hours during which time, the reactor remained shutdown. Prior to restarting the reactor following the leakage increase, the licensee installed a canopy over the top of the valve, moving the pressure retaining surface from the seal ring to the canopy. The modification stopped the leakage and the performance indicator returned to the Green performance level.

Report Details

01 Inspection Scope

This supplemental inspection was performed by the NRC in accordance with Inspection Procedure 95001, Inspection for One or Two White Inputs in a Strategic Performance Area. The inspector reviewed the licensee's root cause evaluation and the corrective actions associated with a White performance indicator in the reactor safety strategic performance area, barrier integrity cornerstone. The reactor coolant system identified leakage performance indicator exceeded the NRC's Reactor Oversight Process threshold of 50 percent of the Crystal River 3, Improved Technical Specification limit (10 gallons per minute) in May 2001. Commensurate with risk, the inspector assessed the adequacy of the licensee's root cause determination and corrective actions. The inspection was completed by review of documents and discussions with licensee personnel. The licensee's nuclear condition report for the leakage and its repair (Condition Report 40706) was reviewed in detail. Some inspection of this issue had been previously completed as discussed in NRC Supplemental Inspection Report 50-302/00-07 and Sections 1R13, 1R17, and 1R20 of NRC Inspection Report 50-302/01-02.

02 Evaluation of Inspection Requirements

02.01 Problem Identification

a. *Determine that the evaluation identifies who (i.e., licensee, self revealing, or NRC), and under what conditions the issue was identified.*

The licensee monitored reactor coolant system leakage as required by Improved Technical Specification (ITS) 3.4.12, "Reactor Coolant System Operational Leakage". The licensee initially found leakage from a decay heat removal system isolation valve (DHV-3) as a result of an evaluation of a change in the unidentified leakage rate and direct observation of leakage from the valve. The valve is used to isolate the decay heat removal system from the reactor and was maintained closed during power operation. The problem of leakage from DHV-3 was entered into the licensee corrective action program when it was found in April 2000. Leakage from the DHV-3 seal ring was considered identified leakage after the direct observations were evaluated and quantified by licensee engineering personnel. Temporary repairs were implemented on the valve which were not fully effective in stopping the leakage. Following a May 2001, plant mode change from power operation to cold shutdown and back to hot shutdown, DHV-3 leakage increased from approximately 2 gallons per minute (gpm) to greater than 5 gpm, the White Performance Indicator threshold. Other contributors to the reactor coolant system identified leak rate, at the time the indicator exceeded the White threshold, were minimal.

The inspector found that the licensee's identification of increased reactor coolant system leakage was appropriate. Increased reactor coolant system leakage, including that from DHV-3, had been previously reviewed by NRC inspectors as documented in NRC Supplemental Inspection Report 50-302/00-07.

b. *Determine that the evaluation documents how long the issue existed, and prior opportunities for identification.*

DHV-3 leakage, initially observed in April, 2000, remained less than two gallons per minute (gpm) until May 21, 2001. Following plant heatup and pressurization (to Mode 3), after a planned shutdown/depressurization (Mode 5), the leakage rapidly increased to greater than five gpm. Licensee personnel anticipated that increased leakage could result from having the valve undergo a pressure cycle. The plant remained shutdown until the valve was modified and the leakage was stopped.

The inspector found the licensee's evaluation of how long the leakage existed, and the identification of the leakage increase to be acceptable.

c. *Determine that the evaluation documents the plant specific risk consequences (as applicable) and compliance concerns associated with the issue.*

Licensee personnel stated that leakage from DHV-3 posed no challenge to the level of safety of the plant. The overall leakage did not exceed technical specification limits and the valve remained capable of performing its decay heat removal and isolation functions.

The inspector found the licensee's risk evaluation of leakage from DHV-3 to be acceptable and no compliance issues were identified.

02.02 <u>Root Cause and Extent of Condition Evaluation</u>

a. *Determine that the problem was evaluated using a systematic method to identify root causes and contributing causes.*

Because DHV-3 was not disassembled, the precise root cause remained unknown. A detailed engineering evaluation of the valve leakage was conducted. The licensee concluded that the two most likely causes of the initial DHV-3 leakage were non-uniform seating forces about the seal ring due to installation error in 1999, or the 1999 plant shutdown and subsequent restart resulted in distorting the pressure seal ring seating force distribution. The May 2001 pressure cycle associated with a planned plant shutdown and heatup caused the increase in leakage.

The inspector found the licensee's engineering evaluation which identified the potential causes of the leakage from DHV-3 to be acceptable.

b. *Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem.*

The licensee assembled summaries of previous seal ring problems at Crystal River 3, as well as industry information, to evaluate the DHV-3 performance effectiveness and reliability. Specific design features for the valve were considered in the licensee's review. The licensee identified that overall reactor coolant system leakage did not exceed technical specification limits and that decay heat removal valve DHV-3, remained capable of performing its decay heat removal and isolation functions.

The inspector found that the licensee's engineering evaluation to be of sufficient detail commensurate with the significance of the problem.

c. *Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience.*

The licensee's reactor coolant system engineer maintained a trend report of reactor coolant system leakage. Significant changes in total, identified, or unidentified leakage on the trend report were evaluated, explained, and reported to management. The licensee monitored reactor coolant system leakage in their maintenance rule program, with leakage monitoring criteria set at the technical specification limits for identified and unidentified leakage. The licensee continued to monitor for reactor coolant leakage from other sources such as reactor coolant system pressure relief valve, RCV-8, and continued corrective actions for a previous White Performance Indicator due to leakage from this valve (See NRC Supplemental Inspection Report 50-302/00-07).

The inspector found the licensee's monitoring and evaluation of reactor coolant system leakage to be acceptable.

d. *Determine that the root cause evaluation included consideration of potential common causes and extent of condition of the problem.*

The licensee completed an evaluation of past seal ring valve leaks for valves similar to DHV-3. The licensee found that, although valves like DHV-3 passed an initial pressurization test following installation/replacement of the seal ring, the subsequent occurrence of leakage was higher than expected for those valves using un-plated nickel seal rings. Some seal ring valves that had developed leaks in the plant's history had been either modified or replaced, while others continued to be monitored. The licensee noted that DHV-3 was unique because it could not be isolated from the reactor and required full core offload for valve disassembly and access to the seal ring.

The inspector found the licensee's evaluation of occurrences of leakage from valves similar to DHV-3 to be acceptable.

02.03 <u>Corrective Actions</u>

a. *Determine that appropriate corrective actions are specified for each root/contributing cause or that there is an evaluation that no actions are necessary.*

The licensee permanently modified DHV-3, which moved the pressure boundary from the seal ring to an installed canopy. As a result of the modification, leakage from DHV-3 stopped. Selected aspects of the modification and its implementation were reviewed by the NRC as documented in Section 1R17 of Inspection Report 50-302/01-02.

The inspector found the licensee's corrective actions for the reactor coolant system leakage were appropriate.

b. *Determine that the corrective actions have been prioritized with consideration of the risk significance and regulatory compliance.*

The inspector found that corrective actions were prioritized and completed so that no operational limits were exceeded.

c. *Determine that a schedule has been established for implementing and completing the corrective actions.*

The licensee evaluated various options for modifying or repairing DHV-3 after the leakage was identified in April 2000. Tracking of corrective actions was done through the licensee's corrective action program. When the leakage reached a level determined by the licensee to be too high, the valve was modified which stopped the leakage. The modification was implemented prior to exceeding any operational limits.

The inspector found that the licensee had planned and scheduled corrective actions appropriately.

d. *Determine that quantitative or qualitative measures of success have been developed for determining the effectiveness of the corrective actions to prevent recurrence.*

The inspector observed that the licensee planned an effectiveness review of the DHV-3 repair following the October 2001 refueling outage. Reactor coolant system integrity continued to be monitored in the licensee's maintenance rule program.

The inspector concluded that these measures were appropriate to determine the effectiveness of the corrective actions.

03 Management Meetings

Exit Meeting Summary

The inspector presented the inspection results to Mr. D. Young, Vice President, Crystal River Nuclear Plant, and other members of licensee management on August 9, 2001. The inspector asked the licensee whether any of the material examined during the inspection should be considered proprietary and no proprietary information was identified.

PERSONS CONTACTED

Florida Power Corporation
P. Saltsman, Senior Systems Engineer
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M. Donovan, Systems Engineering Supervisor
R. Tyree, Nuclear Plant Operations
D. Herrin, Senior Licensing Engineer

NRC
Leonard Wert, Branch Chief, Division of Reactor Projects
S. Sanchez, Resident Inspector, Crystal River 3

DOCUMENTS REVIEWED

Precursor Card 3-C01-1302; Technical Specification 3.4.12, Condition A entered
Nuclear Condition Report 40706; Reactor coolant system leak from DHV-3
Nuclear Condition Report 40522; Reactor coolant identified leakage performance trend
Engineering Trend Report: Reactor coolant system leakage
Reactor Coolant System Quarterly Report, April 2001

ACRONYMS USED

FPC	Florida Power Corporation
DHV	Decay Heat Removal System Isolation Valve
GPM	Gallons Per Minute
ITS	Improved Technical Specifications
RCV	Reactor Coolant System Pressure Relief Valve