September 26, 1988

Docket No. 50-302

DISTRIBUTION
See attached sheet

Mr. W. S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
P. O. Box 219
Crystal River, Florida 32629

Dear Mr. Wilgus:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT RE: SURVEILLANCE INTERVAL FOR REACTOR VESSEL INTERNALS VENT VALVES (TAC NO. 65532)

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TSs) in partial response to your application dated August 14, 1986, as supplemented October 6, 1986 and revised August 2, 1988.

This amendment extends the surveillance interval for reactor vessel internals vent valves from once per 18 months to once per 24 months. Amendments 93 and 94, issued October 21, 1986 and November 7, 1986, respectively, responded to other facets of your original request.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/S/

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 108 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

See previous concurrence*

LA:PDII-2	*PM:PDII-2	EMEB:DEST	*D:PDII-2	OGC-WF
DMiller	HSilver	LMarsh	HBerkow	
08/ /88	08/19/88	08/30/88	08/22/88	09/8/88

9/16 for SER revs.

Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
Generating Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated August 14, 1986, as supplemented October 6, 1986 and revised August 2, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 108, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: September 26, 1988

ATTACHMENT TO LICENSE AMENDMENT NO.108

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by amendment number and contains a vertical line indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

Remove	Insert
3/4 4-32	3/4 4-32

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5:

a. The reactor coolant pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b. of Regulatory Guide 1.14, Revision 1, August 1975.

SURVEILLANCE REQUIREMENTS (Continued)

b. Each internals vent valve shall be demonstrated OPERABLE at least once per 24 months during shutdown, by:

   1. Verifying through visual inspection that the valve body and valve disc exhibit no abnormal degradation,

   2. Verifying the valve is not stuck in an open position, and

   3. Verifying through manual actuation that the valve is fully open with a force of $\leq$425 lbs (applied vertically upward).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

INTRODUCTION

By letter dated August 14, 1986, as supplemented October 6, 1986 and revised August 2, 1988, Florida Power Corporation (FPC, or the licensee) requested an amendment to the Technical Specifications (TS's) appended to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The proposed amendment would extend the surveillance period for inspection and operability testing of reactor vessel internals vent valves (RVVVs) from once per 18 months to once per 24 months.

The original submittal requested an extension of the RVVV surveillance period to once per fuel cycle. Amendment 94, issued November 7, 1986, extended the surveillance period to once per fuel cycle for Cycle 6 only, and noted that the request for permanent change would be addressed as a separate action. The August 2, 1988 submittal revised the permanent request to once per 24 months. This Safety Evaluation (SE) addresses that request on the basis of information provided in the three submittals identified above.

The August 14, 1986 submittal also included a request to extend the surveillance interval for the high pressure injection and low pressure injection pumps and valves to once per fuel cycle for Cycle 6 only. Amendment 93, issued October 21, 1986, responded to that request.

BACKGROUND

Regulatory and Plant Requirements

Under 10 CFR 50.55a(g), the NRC requires inservice testing to verify operational readiness of valves whose function is required for safety. CR-3 TS Sections 4.4.10.b and 4.0.2 require that the RVVVs be demonstrated operable at least once per 18 months, with a provision that an extension of 25% (= 4.5 months) may be granted for the 18-month period. Additionally, the total maximum combined interval for any three consecutive tests may not exceed 3.25 times the 18-month surveillance interval. In order to meet the TS inspection interval requirements, the valve operability at CR-3 should be demonstrated by May 1989.

Licensee's Justification

Due to the length of the last refueling and forced outages at CR-3, the surveillance for the RVVVs is required to be performed prior to the end of the current fuel cycle. Surveillance was not performed on the RVVVs during the forced outages since the surveillance necessitates removal of the reactor vessel head. The reactor vessel head was not removed during the forced outages. The surveillance for the RVVVs is currently required to be completed by May 1989. However, the fuel cycle is not scheduled to end until September 1989.

A permanent change to the surveillance interval will eliminate the necessity for mid-cycle shutdowns to perform this surveillance. The change would also reduce the need for unnecessary removal of the reactor vessel head. This will result in the reduction of unnecessary personnel radiation exposure involved with this evolution as well as a reduction in the probability of a reactor vessel head drop.

Since 1978, the eight RVVVs at CR-3 have each been tested seven times for a total of 56 functional tests without a single failure. The trend of this data parallels that of the other B&W operating reactors. Industry records (from 1973 through the present) indicate that in B&W operating reactors with an approximate total of 95 reactor years of operation, not a single RVVV has failed to demonstrate satisfactory operability in over 480 functional tests and no RVVV has ever stuck open.

The RVVVs are constructed of materials that have satisfactory corrosion resistance to the reactor coolant environment. Tight reactor coolant chemistry controls are also maintained to assure that any corrosion that may occur is insignificant. As a result, it is not considered likely that operability of these valves will be affected by corrosion.

The history of these valves demonstrates they are very reliable. The only degradation of these valves was discovered in November 1978 at two B&W operating reactors. At that time, wear was discovered only at those RVVVs adjacent to the reactor vessel outlet nozzle and only on the RVVVs' locking devices immediately adjacent to the reactor vessel outlet nozzle. However, the valves in this degraded condition were still operational and still capable of performing their intended function. The locking device holds the valve body in place on the core support shield and is not a moving part of the valve. As a result of these flow-induced wear problems associated with the RVVVs, a detailed inspection was conducted at each affected B&W plant site during their next refueling outages. B&W recommended that if wear was discovered on the RVVVs, the locking device of the valve was to be modified. At CR-3, the inspection consisted of a video and ultrasonic testing inspection. Following the RVVV inspection at CR-3 (during the 1980 refueling outage), four of the RVVVs' locking mechanisms were modified to prevent recurrence of the wear problems. Since that time, no further wear indications of the RVVVs have been discovered at CR-3.

DISCUSSION AND EVALUATION

The internals vent valves are installed in the core support shield to prevent a pressure imbalance which might interfere with core cooling following a postulated inlet pipe rupture. The arrangement consists of vent valve assemblies installed in the cylindrical wall of the internal core support shield.

Internals vent valves are included in reactor internals to provide a direct path to the break for steam venting after a loss-of-coolant accident resulting from a postulated cold-leg rupture. The vent valves are required because the arrangement of the RCS can possibly inhibit the venting of steam generated in the core after the system is depressurized if significant quantities of coolant remain in the RCP suction piping at the end of the blowdown period. Without the venting of the steam, a pressure differential would exist between the core region and the reactor vessel internals inlet annulus region, where emergency core coolant is injected, which would prohibit flow into the core. To eliminate the problem, the vent valves are installed in the reactor internals to provide a direct flowpath from the region above the core to the pipe rupture location. The flowpath provides for pressure equalization and permits emergency coolant water to reflood the core rapidly.

The NRC staff took into consideration the record of the past tests of similar valves, which represented about 480 RVVV inspections and exercises at B&W facilities. The information presented by the licensee indicates that RVVVs have demonstrated a high degree of reliability and no failures were found. Recent information has also been submitted by Toledo Edison Company which indicated that the typical span between RVVV inspection and exercise was (with the exception of TMI-1) 12-18 months with a maximum test interval of about 2 years. In the case of TMI-1, the corresponding interval was 37 months. At CR-3, the last surveillance interval, as permitted by Amendment 94, was 29 months. The previous interval was 24 months. As indicated above, there were no failures during these tests.

The NRC staff also evaluated the information pertaining to the RCS chemistry, the compatibility of the mating materials and their corrosion resistance, and the reactor coolant environment. The chemistry of the RCS water is controlled to minimize corrosion, minimize material activation, and maximize the reliability of reactor and steam generator equipment. Comparison of the critical elements of water chemistry such as pH, conductivity, oxygen, chlorides, fluorides, sulfur, and hydrogen for the CR-3 and TMI-1 plants, indicates that the water environment in both plants is similar. Due to the fact that the RVVVs at TMI-1 were not exercised for about 37 months, a length of time greater than that for the RVVVs at CR-3, this comparison is indicative of what would be expected if the request for extension of the testing period for the valves is granted. Corrosion, which could have an effect on the operation of the RVVVs, has been considered. The parts vulnerable to corrosion are the shaft, bushing, and the body, constructed respectively of type 431 martensitic stainless steel, Stellite No. 6, and Type 304 austenitic stainless steel. The corrosion rates of these materials in the RCS hot operating conditions have been verified by the staff in professional literature to be in the range of 0.05 mils/year or less. Because the thickness of the deposit is about three times the rate of corrosion, the expected thickness of the deposited material is 0.15 mils per year. The minimum cold clearance gap dimensions vary from 3 to 60 mils, thus the gap would not become closed such as to hinder the operation of the valve prior to the next test of the RVVVs.

Based on our review as summarized above, we conclude that it would be acceptable to extend the surveillance period of inspection and operability testing of the RVVVs at CR-3 to 24 months.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in surveillance requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 26, 1988

Principal Contributor:

H. Silver

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-72-CRYSTAL RIVER UNIT 3

DATED: September 26, 1988

Docket File
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