DCR 016

November 7, 1986

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
P. O. Box 14042; M.A.C. H-3
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 94 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.

This amendment extends the surveillance interval for reactor vessel internals vent valves (RVVVs) from once per 18 months to once per fuel cycle for Cycle 6 only. You had requested a permanent extension of the surveillance interval for the RVVVs. A permanent change will be addressed as a separate action.

Included in your application was a request to extend the surveillance interval for high pressure injection and low pressure injection pumps and valves from once per 18 months to once per fuel cycle for Cycle 6 only. Amendment No. 93, issued October 21, 1986, responded to that request.

Please note that Section 2.0 of the enclosed Safety Evaluation contains a staff recommendation concerning TS 4.0.2 which you may want to consider in a future TS change request.

Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Brenda Mozafari

Brenda Mozafari, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 94 to DPR-72
2. Safety Evaluation

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cc w/enclosures:
See next page

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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. 94
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, 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. 94 , 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

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 7, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 94

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 vertical lines 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

REACTOR COOLANT SYSTEM

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated August 14, 1986, Reference 1, Florida Power Corporation (FPC or the licensee) requested amendment to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The licensee requested a Technical Specification (TS) change to extend the surveillance period of inspection and operability testing of reactor vessel internals vent valves (RVVVs) to once per fuel cycle. The licensee's request for a revision to the TSs to permanently change the surveillance interval to every refueling outage is still being reviewed. However, to preclude an unnecessary shutdown, this amendment would extend the surveillance interval from once per 18 months to once per fuel cycle for Cycle 6 only but no later than January 1988. The proposed change involves TS 4.4.10.b.

On October 2, 1986, representatives of the Nuclear Regulatory Commission staff and FPC met to discuss additional information regarding this license amendment request. Pursuant to this discussion, the licensee submitted additional information (Reference 2) regarding mechanical design, materials of construction, and chemistry of the reactor coolant system (RCS). On the basis of the information provided by the licensee in the August 14 and Ocotber 6, 1986 submittals, the staff performed an assessment of the safety issues of the subject request.

1.1 Regulatory and Plant Requirements

In 1975, the NRC revised 10 CFR 50.55(a) to require an "Inservice Testing" of various safety related components, including pumps and valves, to be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, "to the extent practical within the limitations of design, geometry, and materials of construction." The 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 time for any three consecutive tests may not exceed 3.25 times the 18-month surveillance interval. In order to meet the latter TS inspection interval requirement, the valve operability at CR-3 should be demonstrated by November 9, 1986.

1.2 Licensee's Justification

Due to the length of the last refueling and the recent reactor coolant pump (RCP) outage at CR-3, the surveillance for the RVVVs will be required to be performed prior to the end of the current fuel cycle. Surveillance was not performed on the RVVVs during this recent outage since the surveillance necessitates removal of the reactor vessel head. The reactor vessel head was not removed during the RCP outage. The surveillance for the RVVVs is currently required to be completed by November 1986. However, the fuel cycle is not scheduled to end until October 1987.

The change in surveillance interval would eliminate the need for shutdown and removal of the reactor vessel head. This will result in minimizing personnel radiation exposure involved with this testing as well as eliminating the probability of a reactor vessel head drop.

Since 1978, the eight RVVVs at CR-3 have each been tested six times for a total of 48 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, in B&W operating reactors with an approximate total of 80 reactor years of operation, not a single RVVV has failed to demonstrate satisfactory operability in over 420 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 device 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 onto the core support shield and is not part of the moving parts 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 on the RVVVs have been discovered at CR-3.

2.0 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 420 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 (Reference 3) 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 two years. In the case of TMI-1, the corresponding interval was 37 months. 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 activations, 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 (Reference 4) 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, we conclude that it would be acceptable to extend the current surveillance period of inspection and operability testing of the RVVVs at CR-3 for fuel cycle 6 only, to coincide with the next reactor head removal but not later than the next refueling outage or January 1988. In addition, TS Section 4.0.2, which requires that the total maximum combined interval time for any three consecutive tests not exceed 3.25 times the 18-month surveillance interval, need not include the current or previous interval times. Therefore, any interval time prior to the next refueling outage will not be required to be considered in meeting this requirement.

3.0 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.

4.0 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: November 7, 1986

Principal Contributors:

G. Hammer
B. Mozafari

References:

1. Letter from Rolf C. Widell, FPC, to Harold R. Denton, NRC, dated August 14, 1986.

2. Letter from Rolf C. Widell, FPC, to Harold R. Denton, NRC, dated October 6, 1986.

3. Letter from Joe Williams, Jr., Toledo Edison Company, to John F. Stolz, NRC, dated June 6, 1986.

4. Uhlig, Herbert, H., "Corrosion and Corrosion Control," John Wiley and Sons Inc., 2nd Edition, 1971.