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CONFORMANCE TO NRR GENERIC LETTER 82-16
CRYSTAL RIVER NUCLEAR GENERATING PLANT UNIT 3

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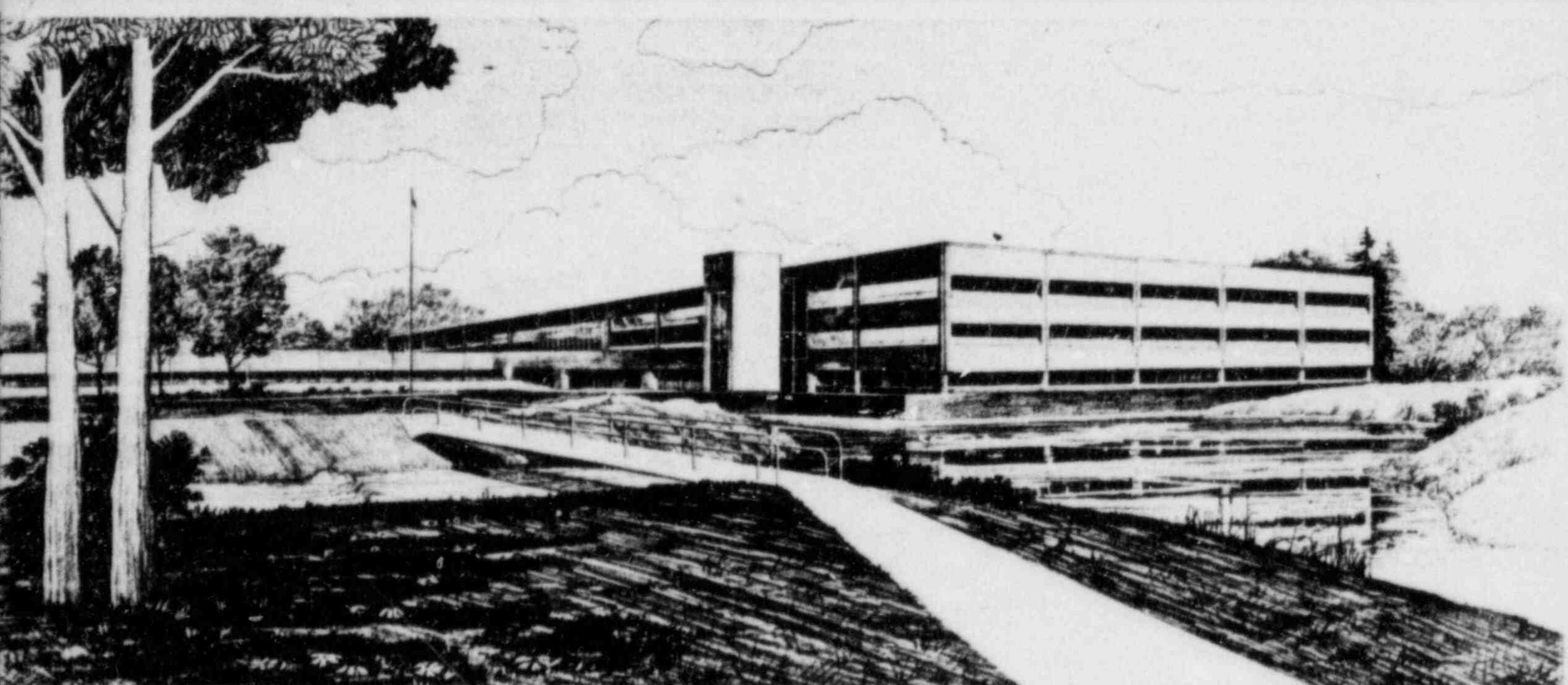

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EG&G Idaho

CONFORMANCE TO NRR GENERIC LETTER 82-16
CRYSTAL RIVER GENERATING PLANT UNIT 3

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ABSTRACT

This EG&G Idaho, Inc., report evaluates the submittal provided by Florida Power Corporation (FPC) for Crystal River Nuclear Generating Plant Unit 3. The submittal is in response to Generic Letter No. 82-16, "NUREG-0737 Technical Specifications (TS)". Applicable sections of the plant's TS are evaluated to determine compliance to the guidelines established in the generic letter.

FOREWORD

This report is supplied as part of the "Technical Assistance for Operating Reactors Licensing Actions" being conducted for the U.S. Nuclear Regulatory Commission Region II by EG&G Idaho, Inc., NRC Licensing Support Section.

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Docket Nos. 50-302
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CONFORMANCE TO NRR GENERIC LETTER 82-16

CRYSTAL RIVER NUCLEAR GENERATING PLANT UNIT 3

1. INTRODUCTION

On September 20, 1982, Generic Letter 82-16[1] was issued by D. G. Eisenhut, Director of Licensing, Office of Nuclear Reactor Regulation (NRR), to all pressurized power reactor licensees. This letter identified a number of items which were required by NUREG-0737[2] to be implemented into the licensee's Technical Specifications (TS) by December 31, 1981. Each licensee was requested to review his facility's TS, to address areas of compliance, and to identify deviations or absence of a specification for the items identified in the generic letter within 90 days of receipt of the letter.

The Florida Power Corporation (FPC), the licensee for Crystal River Nuclear Generating Plant Unit 3 (CR-3), provided a response to the generic letter on September 16, 1983[3].

This report provides an evaluation of the licensee's TS and Nuclear Regulatory Commission (NRC) correspondence with the licensee pertaining to those items identified in the generic letter.

2. REVIEW REQUIREMENTS

The review consists of evaluating the licensee's response, currently approved TS, and other NRR approvals against the criteria set forth in Generic Letter 82-16. The NUREG-0737 items and the criteria established are as follows.

2.1 STA Training (I.A.1.1.3)

The licensee is to address within his TS that a shift technical advisor (STA) to the shift supervisor is provided. In addition, the qualifications, training, and on-duty requirements for the STA should be stated.

2.2 Shift Manning-Overtime Limits (I.A.1.3.1)

The licensee is to provide changes to his TS providing overtime administrative procedure and staffing requirements. The following guidelines were established for the licensee by the NRC.

"a. An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period (all excluding shift turnover time).

c. A break of at least eight hours should be allowed between work periods (including shift turnover time).

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Recognizing that very unusual circumstances may arise requiring deviation from the above guidelines, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management. The paramount consideration in such authorization shall be that significant reductions in the effectiveness of operating personnel would be highly unlikely.

In addition, procedures are encouraged that would allow licensed operators at the controls to be periodically relieved and assigned to other duties away from the control board during their tour of duty."

2.3 Short Term Auxiliary Feedwater System (AFWS) Evaluation (II.E.1.1)

The objective of this item is to improve the reliability and performance of the auxiliary feedwater (AFW) system. TS depend on the results of the licensee's evaluation and the staff review, and are being developed separately for each plant. The limiting conditions of operation (LCO's) and surveillance requirements for the AFW system should be similar to other safety-related systems.[1]

2.4 Safety Grade AFW Initiation and Flow Indication (II.E.1.2)

The AFW system automatic initiation system was to have been control grade by June 1, 1980, and safety grade by July 1, 1981; the AFW system flow indication was to have been control grade by January 1, 1980, and safety grade by July 1, 1981.[1]

2.5 Dedicated Hydrogen Penetrations (II.E.4.1)

Plants that use external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment penetrations dedicated to that service. In satisfying this item, some plants may have to add some additional piping and valves. If so, these valves should be subjected to the requirements of Appendix J of 10CFR 50, and the TS should be modified accordingly.[1]

2.6 Containment Pressure Setpoint (II.E.4.2.5)

The containment pressure setpoint that initiates containment isolation must be reduced to the minimum compatible with normal operating conditions. Most plants provided justification for not changing their setpoint and the NRC has approved their justification by separate

correspondence. The remaining plants must submit a change to the TS with the lower containment pressure setpoint and provide justification if this setpoint is more than 1 psi above maximum expected containment pressure during normal operation.[1]

2.7 Containment Purge Valves (II.E.4.2.6)

Model TS were sent separately to each plant as part of the overall containment purge review. These TS include the requirement that the containment purge valves be locked closed except for safety related activities, verified closed at least every 31 days, and be subjected to leakage rate limits.[1]

2.8 Radiation Signal on Purge Valves (II.E.4.2.7)

The containment purge valves must close promptly to reduce the amount of radiation released outside containment following a release of radioactive materials to containment. TS should include the requirement that at least one radiation monitor that automatically closes the purge valves upon sensing high radiation in the containment atmosphere be operable at all times except cold shutdowns and refueling outages. If not operable, either the plant should begin proceeding to cold shutdown within 24 hours or the purge valves should be closed within 24 hours. Model TS were provided in Standard Technical Specifications format for those plants that are using safety-grade components to satisfy the requirement.[1]

2.9 Upgrade Babcock and Wilcox (B&W) AFWS (II.K.2.8)

Additional long-term AFWS modifications were to be performed in conjunction with Generic Letter 82-16 Items 3 and 4 (2.3 and 2.4 above). The TS implemented for Items 3 and 4 will also address the upgrade of the B&W AFWS; therefore no separate TS would be required for this item for the B&W Plants.

2.10 B&W Safety-Grade Anticipatory Reactor Trip (II.K.2.10)

Safety-grade turbine trip equipment initiating a reactor trip was to be implemented by the B&W designed plants as part of the TMI lessons learned. The licensee is to implement in the TS the trip setpoint, number of channels, trip conditions, minimal channels required for operation, applicable operating modes, actions to be taken, surveillance required and any other requirements for safety-grade equipment.

2.11 B&W Thermal-Mechanical Report (II.K.2.13)

Licensees of B&W operating reactors were required to submit by January 1, 1981, an analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. TS, if required, will be determined following NRC staff review.[1]

2.12 Reporting Safety and Relief Valve Failures and Challenges (II.K.3.3)

NUREG-0660 stated that safety and relief valve failures be reported promptly and challenges be reported annually. The sections of the TS that discuss reporting requirements should be accordingly changed. The NRC has noted that an acceptable alternative would be to report challenges monthly.[1]

2.13 Anticipatory Trip on Turbine Trip (II.K.3.12)

Licensees with Westinghouse-designed operating plants have confirmed that their plants have an anticipatory reactor trip upon turbine trip. Many of these plants already have this trip in the TS. For those that do not, the anticipatory trip should be added to the TS.[1]

B&W Thermal-Mechanical Report, item 2.11 above, is not being evaluated in this report. This item is being handled as an active Three Mile Island (TMI) action item under TAC number 45197. Item 2.13 Anticipatory Trip on Turbine Trip is applicable to Westinghouse designs and therefore is not applicable for CR-3 which is a Babcock and Wilcox design. This item is handled under item 2.10.

3. EVALUATION

The evaluations of Generic Letter 82-16 Items are as follows:

3.1 STA Training (I.A.1.1.3)

The licensee has provided response to NUREG-0737 item I.A.1.1.3 in a letter to the NRC dated January 30, 1982.[5] This letter includes the qualifications, training and duties of the Operational Technical Advisor (OTA).

In Table 6.2-1 of CR-3 TS,[6] the STA is designated as being required in the minimal shift crew composition for operational modes 1, 2, 3, and 4. Section 6.3.1 of the TS states that the OTA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have specific training in plant design and in the response and analysis of the plant for transients and accidents. The exact training program is not in the TS; however, the retraining and replacement program is covered in Section 6.4.1 of the TS.

In a letter from the NRC to FPC dated February 5, 1982,[7] the NRC reviewed FPC's response to NUREG-0737 item I.A.1.1 and evaluated the training program. The NRC concluded that the FPC STA training is acceptable and meets the intent of the guidelines set forth.

Until further guidance is issued by the Commission, no further licensing action is required for this item.

3.2 Shift Manning--Overtime Limits (I.A.1.3.1)

The licensee has stated in his response for CR-3 that the overtime limitations are adequately enforced by administrative procedures and that no amendment to the TS is necessary at this time. This policy was accepted in the Safety Evaluation Report (SER) dated December 7, 1981.[8] The TS for CR-3 does not contain any shift manning overtime limitation requirements.

On June 30, 1983, a notegram from D. C. Fischer, Lead Project Manager for the main topic I.A.1.3[9] was sent to the operating reactor project manager for Crystal River Nuclear Generator Plant Unit 3 requesting that the TAC number be closed out for item I.A.1.3.1 on CR-3. Although NRR has provided acceptance in allowing the licensee to control overtime limitations administratively, it is recommended that this issue be re-examined by NRR under a review separate from Generic Letter 82-16. The basis for this recommendation is that administrative procedures can be changed by the licensee without NRC guidance or approval. Therefore, there is no controlling factor as with the TS and if overtime limitations are not included within the TS, non-compliance to NUREG-0737 Item I.A.1.3.1 can continue to exist.

3.3 Short Term Auxiliary Feedwater System (AFWS) Evaluation (II.E.1.1)

The licensee has stated in his response that Technical Specification Change Request No. 82[10] dated June 22, 1983, addresses NUREG-0737 item II.E.1.1. They also state that FPC is discussing with the NRC Staff further TS changes to improve the Emergency Feedwater System reliability and performance.

Review of the TS Change Request 82 proposed Section 3.7.1.2 for the Emergency Feedwater System indicates that the limiting conditions of operation (LCO's) and surveillance requirements are addressed. These are similar to those of other safety related systems.

Amendment No. 64[11] to the Facility Operating License No. DPR-72 was issued July 12, 1983, providing approval for the Change Request No. 82. The assigned TAC Number for this item is 44668. No further licensing action is required for this item.

3.4 Safety Grade AFW Initiation and Flow Indication (II.E.1.2)

The licensee has stated in his response that the safety-grade Emergency Feedwater Initiation and Control System is scheduled for installation during refuel outage V scheduled for spring of 1985. The licensee states that the associated TS changes will be requested at that time.

The licensee provided a final system description for the planned upgrade of the Emergency Feedwater (EFW) System in a letter to the NRC dated August 11, 1981.[12] A SER was issued September 27, 1982, but was later retracted on November 15, 1982. The assigned TAC Numbers for this item are 49059, 44708, and 44750. We conclude that this item has not met the Generic Letter 82-16 criteria and should be handled as a separate issue.

3.5 Dedicated Hydrogen Penetrations (II.E.4.1)

The licensee has stated in his response that the CR-3 design includes redundant, single failure proof, dedicated hydrogen penetrations; that no additional piping and valves are required to satisfy this item. Because of this, the licensee states that no Amendment to the TS for this item is necessary at this time.

Review of TS Section 3/4.6.1 "Primary Containment", Table 3.6-1, "Containment Isolation Valves" and the surveillance requirements specified for containment integrity (4.6.1.2) indicates that the containment isolation valves are to conform with the criteria specified in Appendix J of 10 CFR 50.

On August 9, 1981,[13] a letter from the NRC to FPC provided acceptance of the existing systems at CR-3 and considered NUREG-0737 item II.E.4.1 closed. The isolation valves for the Hydrogen Purge System are addressed in TS Change Request 82 and approval will be provided in a future Amendment to the TS. No further licensing action will be required for this item.

3.6 Containment Pressure Setpoint (II.E.4.2.5)

The licensee has stated in his response for CR-3, that the NRC provided a SER dated December 10, 1982.[14] The NRC concluded that the present containment pressure setpoint for CR-3 is acceptable. At that time TAC Number 42596 was closed for this item. No further licensing action is required for this item.

3.7 Containment Purge Valve (II.E.4.2.6)

The licensee has stated in his response for CR-3 that the containment purge and vent valves have been closed and that verification that these valves are locked closed when in operating modes 1, 2, 3, and 4 will occur every 31 days.

In a letter from the NRC to FPC dated April 6, 1983,[15] the NRC provided a SER for item II.E.4.2.6 and II.E.4.2 7 and concluded that the requirements for item II.E.4.2.6 have been met. A request was also made by the NRC at that time for FPC to submit a TS request change. On June 22, 1983,[10] FPC submitted Technical Specification Change Request No. 82. This request changes Section 3.6.3.1 of the TS to require the Reactor Building purge supply and exhaust valves to be maintained closed during Modes 1, 2, 3, and 4 and adds surveillance Section 4.6.3.1.3 which requires that these valves are verified closed every 31 days when in Modes 1, 2, 3, and 4. The NRC has issued Amendment 64, which provides approval of Change Request 82. The TS has been updated to satisfy NUREG-0737 item II.E.4.2.6. TAC Numbers 51342 and 42596 will be closed out for this item and no further licensing action will be required.

3.8 Radiation Signal on Purge Valves (II.E.4.2.7)

The licensee has stated in his response for CR-3, that because the containment purge valves are locked closed and verified when in Modes 1, 2, 3, and 4, automatic closing of the valves on high radiation signal is not required and therefore no change to the TS is required at this time.

In a letter from the NRC to FPC dated April 6, 1983,[15] the NRC provided a SER for item II.E.4.2.6 and II.E.4.2.7 and concluded that the requirements for item II.E.4.2.7 have been met. Review of TS Section 4.6.3.1.2.b indicates that on a containment radiation-high test signal, each purge and exhaust automatic valve actuates to its isolation position. The item was completed under TAC Number 42596. No further licensing action is required for this item.

3.9 Upgrade Babcock and Wilcox (B&W) AFWS (II.K.2.8)

The TS implemented for items 3.3 and 3.4 above will also address the upgrade of the B&W AFWS; therefore no separate TS is required. Item 3.3 has been resolved and upon resolution of item 3.4 no further licensing action will be required for this item.

3.10 B&W Safety-Grade Anticipatory Reactor Trip (II.K.2.10)

The licensee has stated in his response for CR-3 that Technical Specification Change Request No. 82 addresses the Safety-Grade Anticipatory Reactor Trip.

Review of Change Request No. 82 indicates that the changes to be made meet the requirements specified in the model TS of Generic Letter 82-16. The change will incorporate into the TS the requirement for a reactor anticipatory trip on the trip of both main feedwater pumps and the trip of the main turbine.

In the letter issued by the NRC to FPC dated November 2, 1981,[16] the NRC provided a SER and concluded that the proposed modifications for the anticipatory reactor trips to be acceptable. This item, upon issuing of the Amendment, will meet the requirements of Generic Letter 82-16 and close out TAC Number 51342. No further licensing action will then be required.

3.11 B&W Thermal-Mechanical Report (II.K.2.13)

The Thermal-Mechanical Report for CR-3 is being handled as an active TMI action item under TAC number 45197. No licensing action is required by Generic Letter 82-16 for this item.

3.12 Reporting Safety and Relief Valve Failures and Challenges (II.K.3.3)

The licensee has stated in his response for CR-3 that FPC has committed to report challenges to the valves in the CR-3 annual report as required by TS Section 6.9.1.8.b. Therefore, the requirements of NUREG-0737 item II.K.3.3 have been implemented and no change to the TS is required at this time.

Sections 6.9.1.4 and 6.9.1.8.b of the TS dealing with annual reports and prompt notification do not specifically address the reporting of the challenges to the Safety and Relief Valves. The present TS for CR-3 does not comply with NUREG-0737 item II.K.3.3 and the Generic Letter 82-16 requirements. The assigned TAC Number for this item is 45333. Further licensing action will be required for this item.

3.13 Anticipatory Trip on Turbine Trip (II.K.3.12)

The licensee has stated in his response for Crystal River Nuclear Generator Plant Unit 3 that although not a Westinghouse design, the TS adequately cover the anticipatory trip on turbine trip.

Review of the TS Tables 2.2-1, 3.3-1 and 3.3-2 indicate that the requirements set forth in Generic Letter 82-16 have been met. No further licensing action is required.

4. CONCLUSIONS

Based on our review, we find the licensee conforms to those issues addressed in Generic Letter 82-16 on TS, except for those identified as follows:

1. Section 3.1 STA Training--Until further guidance is provided by the Commission, no further licensing action can be taken to determine whether the exact training program for the STA is required to be in the TS.

2. Section 3.2 Shift Manning-Overtime Limits--The TS for CR-3 does not contain shift-manning overtime limits, however, FPC's overtime limits policy was accepted by the NRC. However, it is recommended that this issue be re-examined by NRR under a review separate from Generic Letter 82-16.

3. Section 3.4 Safety-Grade AFW System Initiation and Flow Indication--The safety-grade emergency feedwater and control system is scheduled for installation during refuel outage V scheduled for the spring of 1985. The licensee has committed to change the TS in compliance to NUREG-0737 item II.E.1.2 and Generic Letter 82-16.

4. Section 3.5 Dedicated Hydrogen Penetrations--The isolation valves for the Hydrogen purge system are not included in the present TS, but are addressed in the licensee's Change Request No. 82. It is expected that the NRC will issue an Amendment change approval for addition of these valves to the TS at a future date.

5. Section 3.9 Upgrade Babcock and Wilcox (B&W) AFWS--For this item to be acceptable and meet the Generic letter 82-16 requirements, item 3.4 above will need to be accepted.

6. Section 3.10 B&W Safety-Grade Anticipatory Reactor Trip--This item will be acceptable upon issuance of an Amendment to the TS.

7. Section 3.12 Reporting Safety and Relief Valve Failures and Challenges--The TS for CR-3 does not comply with the requirements of NUREG-0737 item II.K.3.3 which specifies including in the TS that failures and challenges be listed to be reported annually, nor does the TS comply with the model TS of Generic Letter 82-16.

5. REFERENCES

1. D. G. Eisenhut, NRC letter to All Pressurized Power Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter 82-16)," September 20, 1982.

2. NUREG-0737 Clarification of TMI Action Plan Requirements published by the Division of Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, November 1980.

3. G. R. Westafer, Florida Power Corporation letter to J. F. Stolz, Director of Nuclear Reactor Regulation, "Crystal River Unit 3 Docket No. 50-302, Operating License No. DPR-72, Generic Letter 82-16, NUREG-0737 Technical Specifications," September 16, 1983.

4. D. G. Eisenhut, NRC letter to All Licensees of Operating Plants, Applicants for an Operating License, and Holders of Construction Permits, "Nuclear Power Plant Staff Working Hours (Generic Letter No. 82-12)," June 15, 1982.

5. P. Y. Baynard, Florida Power Corporation letter to D. G. Eisenhut, Office of Nuclear Reactor Regulation, "Crystal River Unit 3, Docket No. 50-302, Operating License No. DPR-72, NUREG-0737, Post-TMI Requirements," January 30, 1981.

6. Crystal River Nuclear Generating Plant Unit 3 Technical Specifications, Appendix "A" to License No. DPR-72, December 3, 1976, Amendment 66.

7. J. F. Stolz, NRC letter to J. A. Hancock, Florida Power Corporation, "NUREG-0737 item I.A.1.1 Shift Technical Advisor (STA)," February 5, 1982.

8. J. F. Stolz, NRC letter to J. A. Hancock, Florida Power Corporation, "TMI Action Plan Items I.A.1.3.1, I.C.5, and I.C.6 as described in NUREG-0737," December 7, 1981.

9. D. C. Fischer, Lead Project Manager for I.A.1.3, notegram to all operating reactor project managers, "TAC Closeout," June 30, 1983.

10. G. R. Westafer, Florida Power Corporation letter to H. R. Denten, Office of Nuclear Reactor Regulation, "Crystal River Unit 3, Docket No. 50-302, Operating License No. DPR-72, Technical Specification Change Request No. 82," June 22, 1983.

11. J. F. Stolz, NRC letter to W. S. Wilgus, Florida Power Corporation, "Amendment No. 64 to Facility Operating License No. DPR-72 for Crystal River Unit 3," July 12, 1983.

12. Florida Power Corporation letter to J. F. Stolz, Office of Nuclear Reactor Regulation, "Crystal River Unit 3, Docket No. 50-302, Operating License No. DPR-72, NUREG-0737 Item II.E.1.2 Emergency Feedwater System Upgrade," August 11, 1981.

13. J. F. Stolz, NRC letter to J. A. Hancock, Florida Power Corporation, "Task Action Plan NUREG-0737, II.E.4.1.1 and II.E.4.1.2 Dedicated Hydrogen Penetrations," August 9, 1981.

14. J. F. Stolz, NRC letter to J. A. Hancock, Florida Power Corporation, "Crystal River Unit 3 (CR-3), NUREG-0737, Item II.E.4.2 Position 5," December 10, 1982.

15. J. F. Stolz, NRC letter to W. S. Wilgus, Florida Power Corporation, "Completion of Review of NUREG-0737, Items II.E.4.2.6 and II.E.4.2.7 for Crystal River Unit No. 3 (CR-3)," April 6, 1983.

16. J. F. Stolz, NRC letter to J. A. Hancock, Florida Power Corporation, "Crystal River Unit 3 Safety Grade Anticipatory Reactor Trips (ARTs), NUREG-0737, Item II.K.2.10," November 2, 1981.

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16. ABSTRACT *(200 words or less)*

This EG&G Idaho, Inc. report evaluates whether the designated Operating Reactor Plant has conformed to the requirements of the NRR Generic Letter No. 82-16, "NUREG-0737 Technical Specifications."

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WASHINGTON, D. C. 20555

ENCLOSURE 2
SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT TO THE FACILITY OPERATING LICENSE NO. DPR-72
FLORIDA POWER CORPORATION
CRYSTAL RIVER NUCLEAR GENERATING UNIT 3
DOCKET NO. 50-302

INTRODUCTION AND BACKGROUND

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of Technical Specifications for each of the eleven items in Generic Letter 83-37, which was issued to all Pressurized Water Reactor (PWR) licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

1. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the Generic Letter, and

2. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

Florida Power Corporation (FPC), the licensee for Crystal River 3 (CR3), has proposed Technical Specification changes to a number of GL 83-37 items in two major submittals, one dated January 17, 1983 and the other dated June 22, 1983. A number of supplemental submittals revised the proposed Technical Specification changes have been reviewed by the staff and the results of that review are documented in this evaluation report.

Evaluations and Conclusions

Our evaluations and conclusions regarding licensee conformance with the guidance of GL 83-37 for each of the eleven items are covered in the following subparagraphs:

1. Reactor Coolant System Vents (II.B.1)

   GL 83-37 stated that at least one reactor coolant system vent path (consisting of at least two valves in series which are powered from emergency buses) shall be operable and closed at all times (except for cold shutdown and refueling) at each of the following locations:

   a. Reactor Vessel Head
   b. Pressurizer steam space
   c. Reactor coolant system high point

FPC submitted proposed Technical Specification (TS) for the pressurizer and hot leg (high point) vents on June 22, 1983 and submitted a modified version of the proposed TS on February 24, 1984, following receipt of GL 83-37. In a separate licensing action, the staff granted a schedular exemption on July 21, 1983 which allowed deferral of installing a vent in the reactor vessel head at Crystal River Unit 3 until the first outage of sufficient duration after December 31, 1985. The proposed TS would add a new section 3/4.4.11 to provide Limiting Conditions for Operation (LCO's) and Surveillance Requirements for the vent paths which have been installed per NUREG-0737, Item II.B.1. The staff issued a Safety Evaluation on September 8, 1983, stating the CR3 vent system design was found to be acceptable. The TS's proposed by FPC have been evaluated using the guidance of GL 83-37 and are consistent with our guidance with the following exceptions:

a) The GL 83-37 Model TS's specify that, when one of the three RCS vent paths becomes inoperable, the path should be returned to operable status within 30 days or the plant shut down to COLD SHUTDOWN. This provision has been omitted in the TS's proposed by FPC with the justification that deletion will not increase the probability or consequences of an accident. We have evaluated this deviation from the guidance and determined that the licensee should have provisions to return the inoperable vent path to operable status within 30 days as specified by the staff. This provision will ensure adequate venting capability during an accident situation.

b) The proposed interval in section 4.4.11 for performing operability surveillance was specified as once per refueling cycle. To be consistent with other sections of the CR3 TS's and with the Model TS's, this interval should be stated as once per 18 months.

c) The model TS's have a surveillance requirement (section 4.4.11.3) to verify flow through the reactor coolant vent paths during venting once per 18 months during mode 5 or 6. This requirement was not included in the TS's proposed by FPC on the basis that the RCS would have to be pressurized to verify correct flow. The intent is to verify that a flow path does exist and not to verify flow magnitude. The staff will require inclusion of this surveillance in the CR3 TS's.

The TS's for this item should be resubmitted with the changes discussed above.

2. Postaccident Sampling (II.B.3)

Licensees should ensure that their plant has the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. An administrative program should be established, implemented and maintained to ensure this capability. The program should include:

a) training of personnel
b) procedures for sampling and analysis, and
c) provisions for maintenance of sampling and analysis equipment

It is acceptable to the Staff if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operation manuals. A copy of the program should be easily available to the operating staff during accident and transient conditions.

FPC has not submitted a proposed TS for this item similar to model TS 6.8.4 in GL 83-37. Such a TS should be submitted for inclusion into the administrative section of the CR3 administrative TS's.

3. Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)

The objective of this item is to improve the reliability and performance of the auxiliary feedwater (AFW) system. Technical Specifications depend on the results of the licensee's evaluation and staff review of each plant. Guidance for the limiting conditions of operation (LCO) and surveillance requirements for the AFW system was given in GL 83-37. The staff has reviewed the present CR3 TS's and found them to be in essential agreement with the guidance, considering that CR3 has two AFW pumps and not three. The guidance, however, suggested a surveillance item 4.7.1.3 applicable to CR3 which would require a flowpath verification after any refueling outage or other cold shutdown of longer than 30 days. This is not presently in the CR3 TS's. The staff issued a Safety Evaluation of Item II.E.1.1 on May 1, 1984, which documented the fact that FPC has committed to various plant modifications and a technical specification change as part of a project to upgrade the AFW system. The TS which FPC has committed to will require a flow test of the emergency feedwater system following extended cold shutdowns. The staff, upon submittal of the additional TS, considers the item acceptable.

4. Noble Gas Effluent Monitors (II.F.1.1)

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be operable at all times during plant operation, but they are not required for safe shutdown of the plant. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability in a reasonable period of time. Considering the importance of the availability of the equipment and possible delays involved in administrative controls, 7 days is considered to be the appropriate time period to restore the operability of the monitor. An alternate method for monitoring the

effluent should be initiated as soon as practical, but no later than 72 hours after the identification of the failure of the monitor. If the monitor is not restored to operable conditions within 7 days after the failure a special report should be submitted to the NRC within 14 days following the event, outlining the cause of inoperability, actions taken and the planned schedule for restoring the system to operable status.

FPC has supplemented the existing normal range monitors to provide noble gas monitoring in accordance with Item II.F.1.1. Proposed TS's were submitted that are consistent with the guidelines provided in our Generic Letter 83-37. We conclude that the proposed TS's for Item II.F.1.1 are acceptable. These will be issued as a part of the Radioactive Effluent Technical Specification (RETS) amendment.

5. <u>Sampling and Analysis of Plant Effluents</u> (II.F.1.2)

Each operating nuclear power reactor should have the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. An administrative program should be established, implemented and maintained to ensure this capability. The program should include:

a) training of personnel
b) procedures for sampling and analysis, and
c) provisions for maintenance of sampling and analysis equipment

It is acceptable to the staff if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operation manuals. A copy of the program should be readily available to the operating staff during accident and transient conditions.

As discussed in Item 2. above, FPC should submit a TS in accordance with GL 83-37 on postaccident sampling capability.

6. <u>Containment High-Range Radiation Monitor</u> (II.F.1.3)

GL 83-37 states that a minimum of two in-containment radiation-level monitors with a maximum range of 10^8 rad/hr (10 R/hr for photon only) should be operable at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible. If the monitor is not restored to operable condition within 7 days after the failure, a special report should be submitted to the NRC within 14 days following the event, outlining the cause of inoperability, actions taken and the planned schedule for restoring the equipment to operable status.

FPC has installed two in-containment monitors at CR3,which is consistent with the guidance of TMI Action Plan Item II.F.1.3. GL 83-37 provided guidance for limiting conditions of operation and surveillance requirements for these monitors. The licensee proposed TS's that are consistent with the guidance provided in GL 83-37. We conclude that the proposed TS's for Item II.F.1.3 are acceptable.

7. Containment Pressure Monitor (II.F.1.4)

Containment pressure should be continuously indicated in the control room during Power Operation, Startup and Hot Standby modes of operation. Two channels should be operable at all times when the reactor is operating in any of the above mentioned modes. TS's for these monitors should be included with other accident monitoring instrumentation and Limiting Conditions for Operation for the containment pressure monitor should be similar to other accident monitoring instrumentation included in the present TS's. Acceptable LCO and surveillance requirements for accident monitoring instrumentation were included GL 83-37.

Crystal River Unit 3 has been provided with two supplementary channels for monitoring containment pressure following an accident. FPC has proposed TS's that are consistent with the guidelines contained in 83-37. We conclude that the proposed TS's for containment pressure monitor are acceptable.

8. Containment Water Level Monitor (II.F.1.5)

The guidance provided in GL 83-37 stated that a continuous indication of containment water level should be provided in the control room during Power Operation, Startup and Hot Standby modes of operation. TS's for the containment water level monitors should be included with other accident monitoring instrumentation in the present TS's and LCO's for the wide range monitors should be similar to other accident monitoring instrumentation. Acceptable LCO and surveillance requirements for accident monitoring instrumentation were included in GS 83-37.

The Crystal River Unit 3 Reactor Building water level monitors provide the capability required by TMI Action Plan Item II.F.1.5 The proposed TS's contain Limiting Conditions of Operation and surveillance requirements that are consistent with the guidance contained in GL 83-37. We conclude that the proposed TS's for containment (Reactor Building) water level monitors are acceptable.

9. Containment Hydrogen Monitor (II.F.1.6)

GL 83-37 stated that two dependent containment hydrogen monitors should be operable at all times when the reactor is operating in Power Operation or Startup modes. LCO's for these monitors should include the requirement that with one hydrogen monitor inoperable, the monitor should be restored to operable status within 30 days or the plant should be brought to at least a hot standby condition within the next 6 hours. If both monitors are inoperable, at least one monitor should be restored to operable status within 72 hours or the plant should be brought to at least hot standby condition within the next 6 hours.

FPC proposed TS's on this item in the June 22, 1983 submittal. Revised TS requirements were submitted on February 24, 1984 to change the LCO and surveillance requirements. We have evaluated the TS's proposed by FPC as well as the justification provided for deviations from the model TS's in GL 83-37.

The CR3 containment hydrogen monitors are normally operable but not in service during plant operation (i.e. not continuously sampling the containment environment). NUREG-0737 only requires these monitors to be capable of providing indication within 30 minutes of the initiation of safety injection. The piping configuration for these monitors is such that no test or calibration connections exist which would allow test gas required for calibration to be collected prior to being returned to the containment. Therefore, any combustible gas used for the channel calibration (as specified in the model TS's once each 92 days) would be vented directly into the containment, which is undesirable during plant operation. Furthermore, to line the hydrogen monitor up to perform the channel calibration would require opening containment isolation valves which, during plant operation, are required elsewhere in the CR3 TS's to be locked closed. Therefore, FPC has proposed no channel checks, since the monitors are not in continuous operation, and no channel calibration every 92 days, since this would conflict with other TS's and would vent combustible gases into the containment during plant operation. In lieu of these tests, FPC has proposed a channel functional test at least once per 31 days and an accuracy verification, using test gas, once each refueling cycle during plant shutdown. The staff considers that once per refueling cycle means once each 18 months. Based on as-built plant-specific configurations, the staff finds that these deviations have been adequately justified. We conclude that the proposed TS's for containment (Reactor Building) hydrogen monitors are acceptable as proposed in FPC's February 24, 1984 submittal. However, FPC should commit to providing test connections, as required, which would allow channel calibration on a more frequent basis as intended by GL 83-37.

10. Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

Subcooling margin monitors, core exit thermocouples, and a reactor coolant inventory tracking system (e.g., differential pressure measurement system) may be used to provide indication of the approach to, existence of, and recovery from inadequate core cooling (ICC). This instrumentation should be operable during Power Operation, Startup, and Hot Shutdown modes of operation for each reactor. TS's for exit thermocouples and the reactor coolant inventory tracking system should be included with other accident monitoring instrumentation. Four core-exit thermocouples in each core quadrant and two channels in the reactor coolant tracking system are required to be operable when the reactor is operating in any of the above mentioned modes. A minimum of two core-exit thermocouples in each quadrant and one channel in the reactor coolant tracking system should be operable at all times when the reactor is operating in any of the above mentioned modes. Typical acceptable LCO and surveillance requirements for accident monitoring instrumentation were provided in GL 83-37.

Crystal River Unit 3 presently has two subcooling margin monitors (SMM's) and 52 core exit thermocouples (CET's) which can be used to detect inadequate core cooling. Both SMM's and their 12 associated CET's are powered from Class 1E power supplies. The licensee has committed to installing an instrumentation system which fully meets NUREG-0737 Item II.F.2 during the next refueling outage, presently scheduled to start in

March 1985. The proposed system would: upgrade the core exit thermocouples to provide at least four qualified CET's in each core quadrant; add differential pressure instrumentation to monitor coolant inventory in the full height of the two hot legs and in the reactor vessel head area when reactor coolant pumps are not operating; and add reactor coolant pump power and inlet temperature instrumentation to track coolant inventory when the pumps are running. FPC will submit proposed TS changes for the upgraded ICC instrumentation in conjunction with the upcoming refueling outage. Thus, the staff considers review of this GL 83-37 item to be an open item pending receipt of proposed TS's.

11. Control Room Habitability Requirements (III.D.3.4)

Licensees should assure that control room operators will be adequately protected against the effects of the accidental release of toxic and/or radioactive gases and that the nuclear power plant can be safety operated or shut down under design basis accident conditions. If the results of the analyses of postulated accidental release of toxic gases (at or near the plant) indicate any need for installing the toxic gas detection system, it should be included in the TS's. Acceptable LCO and surveillance requirements for such a detection system (e.g., chlorine detection system) are provided in GL 83-37. The control room habitability requirements should also be included in the TS's for the control room emergency air cleanup system. Two independent control room emergency air cleanup systems should be operable continuously during all modes of plant operation and capable of meeting design requirements.

FPC submitted proposed TS's on June 22, 1983 to provide LOC's and surveillance requirements for the chlorine, sulfur dioxide and ammonia detectors installed per NUREG-0737 Item III.D.3.4. The proposed action statements require operation of the control room emergency ventilation system in the recirculation mode in the event of inoperable toxic gas detection system instruments. The requirements are consistent with the guidelines in GL 83-37. We conclude that the proposed TS's for control room habitability are acceptable.

ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of a facility component located within the restricted area. The staff has determined that the amendment involves no significant increase in the amounts 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 discussion above, that (1) there is reasonable assurance that the health, safety and interest 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 1, 1984

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