September 17, 2001

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

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM AND CONTROL ROOM VENTILATION SYSTEM (TAC NO. MB0241)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 199 to Facility Operating License No. DPR-72 for the Crystal River Unit 3. The amendment consists of changes to the existing Technical Specifications (TS) in response to your letter dated October 3, 2000, as supplemented June 14, August 28, and September 7, 2001. The amendment adopts a full implementation of the alternative source term (AST) and conforms to the intent of Technical Specification Task Force Traveler 287, and revises TS 3.3.16, "Control Room Isolation - High Radiation"; 3.7.12, "Control Room Emergency Ventilation System"; 3.7.18, "Control Complex Cooling System"; and 5.6.2.12, "Ventilation Filter Testing Program (VFTP)." A new Section 5.6.2.21, "Control Complex Habitability Envelope Program" has been added. The full implementation of the AST replaces the current accident source term used in design basis radiological analyses with an AST pursuant to Title 10, *Code of Federal Regulations*, Section 50.67, "Accident Source Term." The technical specification changes are based on the results of revised offsite and control room dose calculations for the CR-3 design basis accidents using the proposed AST.

Upon approval of these changes, Florida Power Corporation (FPC) will close out the justification for continued operation described in FPC letter dated January 14, 1998. The NRC requested additional information by letter dated May 24, 2001, and FPC responded to this request by letter dated June 14, 2001.

Since the basis of the amendment request is now TSTF-287, FPC's responses to Request for Information Questions 7 and 8 and Attachment E (Design Calculation Number M97-0137) in FPC's letter to the NRC dated June 14, 2001, are no longer applicable. In telephone correspondence between the NRC and FPC on August 22, 2001, FPC also committed to remove CR-3 Design Calculation Number M97-0137, Revision 4 as an input to the Control Room Habitability Report and from the CR-3 breach margin control program. Additionally, Regulatory commitments have been made by FPC in the October 3, 2000, and June 14, 2001 letters to the NRC.

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

Sincerely,

***/RA/***

John M. Goshen, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 199 to DPR-72
2. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

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SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 199
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 October 3, 2000, as supplemented June 14, August 28, and September 7, 2001, 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. 199, 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: September 17, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 199

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

| Remove | Insert |
|---|---|
| 3.3-36 | 3.3-36 |
| 3.3-37 | 3.3-37 |
| B3.3-119 | B3.3-119 |
| B3.3-120 | B3.3-120 |
| B3.3-121 | B3.3-121 |
| B3.3-122 | B3.3-122 |
| B3.3-123 | B3.3-123 |
| 3.7-24 | 3.7-24 |
| 3.7-25 | 3.7-25 |
| 3.7-26 | 3.7-26 |
| 3.7-37 | 3.7-37 |
| 3.7-38 | 3.7-38 |
| B3.7-60 | B3.7-60 |
| B3.7-61 | B3.7-61 |
| B3.7-62 | B3.7-62 |
| B3.7-63 | B3.7-63 |
| B3.7-64 | B3.7-64 |
| B3.7-65 | B3.7-65 |
| B3.7-65A | B3.7-65A |
| B3.7-65B | B3.7-65B |
| B3.7-86 | B3.7-86 |
| B3.7-87 | B3.7-87 |
| 5.0-18 | 5.0-18 |
| 5.0-19 | 5.0-19 |
| 5.0-20 | 5.0-20 |
| 5.0-23A | 5.0-23A |
| ------ | 5.0-23B |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 199 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 October 3, 2000, as supplemented June 14, August 28, and September 7, 2001, Florida Power Corporation (FPC, the licensee) submitted a request for changes to the Crystal River Unit 3 (CR-3) Technical Specifications (TS) and associated Bases pages. The NRC requested additional information by letter dated May 24, 2001. Florida Power Corporation (FPC) responded to this request by letter dated June 14, 2001. The amendment adopts a full implementation of the alternative source term (AST) and Technical Specification (TS) Task Force Traveler (TSTF)-287. It revises TS 3.3.16, "Control Room Isolation - High Radiation"; 3.7.12, "Control Room Emergency Ventilation System"; 3.7.18, "Control Complex Cooling System"; and 5.6.2.12, "Ventilation Filter Testing Program (VFTP)." A new Section 5.6.2.21, "Control Complex Habitability Envelope (CCHE)" Program has been added. TS 3.7.12 has been revised to establish actions to be taken for an inoperable control room emergency ventilation system (CREVS) due to a degraded CCHE. The changes allow up to 24 hours to restore the CCHE to operable status when two CREVS trains are inoperable due to an inoperable CCHE in MODES 1, 2, and 3. In addition, a Limiting Condition for Operation (LCO) Note is added to allow the CCHE to be opened under administrative control without affecting the CREVS operability. Surveillance Requirement (SR) 3.7.12.4 has been revised to verify the CCHE is maintained in accordance with the CCHE Integrity Program. The applicable TS Bases have been revised to document the TS changes and to provide supporting information. On approval of these amendments by the U.S. Nuclear Regulatory Commission (NRC), FPC will close out the justification for continued operation (JCO) described in the FPC letter dated January 14, 1998. The June 14, August 28, and September 7, 2001, letters provided clarifying information that did not change the conclusions of the proposed no significant hazards consideration finding of the original *Federal Register* notice.

## 2.0 BACKGROUND

Title 10, *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC)-19 establishes the requirement that the control room be able to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Specifically, licensees using an alternative source term under 10 CFR 50.67 must meet the requirements of 10 CFR Part 50, Appendix A, GDC-19 "except that with regard to control room access and occupancy, adequate protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in section 50.2 for the duration of the accident." The TS amendment is based, in part, on the results of revised offsite and control room dose calculations for the Crystal River Unit 3 (CR-3) design basis accidents (DBA) using the AST methodology.

Existing SR 3.7.12.4, which tests the integrity of the control building boundary, requires the satisfactory performance of an integrated leakage test. While other surveillance requirements in the same specification test the operability and function of the ventilation train, the integrated leakage test ensures that the control boundary leak tightness is adequate to meet the design assumptions for the post-accident operator doses.

Currently, when the surveillance is not met, the required action is to restore the CCHE boundary within 7 days. However, this action and the associated surveillance would not be valid after Fuel Cycle 12. There would be no corresponding Conditions, Required Actions, or Completion Times specified in LCO 3.7.12 should the control room boundary become inoperable. Under the existing specifications this situation results in both trains being declared inoperable and LCO 3.0.3 must be entered. Requiring the plant to enter LCO 3.0.3 when the CCHE boundary is not intact does not provide time to affect required repairs or corrective maintenance activities and is inconsistent with TSTF-287.

The proposed change (refer to 2(f) below) is similar in nature to Standard TS LCOs used for the secondary containment (boiling water reactors) and shield building (pressurized water reactors) which allows 24 hours to restore the secondary containment or shield building envelope to operable status before requiring an orderly shutdown from operating conditions.
The amendment required re-evaluations by FPC of certain analyses in the Control Room Habitability Report including the analysis for the protection of the control room operators in the event of a toxic chemical release accident. Additionally, the change in Equipment Qualification doses due to adoption of the AST required evaluation. This amendment is consistent with NUREG-1430, "Standardized Technical Specifications for Babcock and Wilcox Plants," and TSTF-287.

Specifically, the amendment:

1. Replaces the current accident source term used in design basis radiological analyses with an AST pursuant to 10 CFR 50.67 and the guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." FPC proposed a full implementation of the AST as described in Regulatory Guide (RG) 1.183.

2. Reflects updated the Final Safety Analysis Report (FSAR) Chapter 14 analyses of offsite and control room radiological consequences of the DBAs to support the AST implementation. These updates address the AST and the TEDE criteria in 10 CFR 50.67 using the guidance of RG 1.183, except where noted in this evaluation.

3. Proposes revisions to ITS 3.7.12, “Control Room Emergency Ventilation System (CREVS),”

    a. A Note has been added to LCO 3.7.12 for the CREVS to allow the CCHE to be opened intermittently under administrative control.

    b. Revises LCO 3.7.12 Condition B to specify that 24 hours are allowed to restore an inoperable CCHE to operable status. All other Conditions have been administratively re-labeled to support the change.

    c. Revises LCO 3.7.12 Condition D such that two inoperable CREVS trains (for reasons other than CCHE breach) will require shut down.

    d. Removes applicability of the ITS during movement of irradiated fuel assemblies. This change impacts the fuel handling accident analysis.

    e. Removes the LCO Conditions specific to movement of irradiated fuel.

    f. A revised SR 3.7.12.4 has been added to TS 3.7.12 to verify the CCHE is maintained in accordance with the Control Complex Habitability Envelope Integrity Program. The associated Bases for SR 3.7.12.4 are revised accordingly.

    g. Provides miscellaneous conforming editorial changes.

4. Implements changes to ITS 5.6.2.12, “Ventilation Filter Testing Program,” to:

    a. Remove the auxiliary building ventilation exhaust system (ABVES) from the program.

    b. Change the methyl iodine penetration acceptance criteria for control room filters from “less than 2.5%” to “less than 5.0%.”

5. Deletes ITS 3.3.16, “Control Room Isolation - High Radiation.”

6. Implements changes to ITS 3.7.18, “Control Complex Cooling System,” to remove applicability of the ITS during movement of irradiated fuel assemblies.

7. Adds ITS 5.6.2.21, “Control Complex Habitability Envelope Program.”

8. Implements changes to ITS Bases 3.7.12 and 3.7.18.

## 3.0 EVALUATION

### 3.1 Alternative Source Term Evaluations

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, “Accident Source Term,” which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their design basis accident analyses with ASTs.

Regulatory guidance for the implementation of these ASTs is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Section 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. FPC's application of October 3, 2000, addresses these requirements in proposing to use the AST as the CR-3 DBA source term used to evaluate the radiological consequences of DBAs. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC-19, for the DBAs in the CR-3 licensing basis. This review included the analysis for protection of the control room operators in the event of a toxic chemical release accident. The NRC reviewed the FPC implementation of the AST and TEDE criteria and determined it meets the requirements of 10 CFR 50.67 and the guidance provided in RG 1.183. It, therefore, is acceptable.

3.2 Radiological Consequences of DBAs Evaluation

The NRC reviewed the licensee's analysis methods, assumptions, and inputs using docketed information provided by the licensee, including the FSAR. Although the NRC performed independent calculations as a means of confirming the licensee's results, the NRC's acceptance is based on the licensee's analysis. FPC used the NRC-sponsored RADTRAD computer code for performing a large portion of their re-analyses. The NRC's review was facilitated by the detailed descriptions of the RADTRAD modeling and inputs provided in the FPC application. FPC's analyses calculated the offsite and control room doses resulting from the following postulated accidents:

-main steam line break (MSLB)
-steam generator tube rupture (SGTR)
-fuel handling accident (FHA)
-control rod ejection accident (CRE)
-loss-of-coolant accident (LOCA)
-letdown line break accident (LLA)
-waste gas decay tank rupture (WGDT)

The locked rotor accident, typically addressed in pressurized-water reactor safety analyses, is not projected to result in core damage at CR-3. As such, a radiological analysis is not necessary for this event.

FPC updated many analysis assumptions for consistency with the AST regulatory guidance. However, FPC proposed retaining some aspects (e.g., no consideration of iodine spiking or loss-of-offsite power (LOOP)) of the existing licensing basis, particularly for offsite dose calculations. As stated in RG 1.183, the NRC believes that it is preferable for both the offsite and control room analyses to use assumptions consistent with RG 1.183. However, prior design bases that are unrelated to the AST, or are unaffected by the AST, may continue as a facility's design basis, provided that no new or unreviewed safety issues are created by the use of these assumptions simultaneously with the implementation of the AST. These exceptions will be addressed further in the discussion for the affected accident analysis.

For the control room habitability analyses, FPC performed analyses for the SGTR and LLA that incorporated assumptions for iodine spiking and LOOP. However, consistent with a protocol adopted in an earlier amendment request dated July 30, 1998, FPC has asserted that these analyses are not part of the licensing basis for CR-3 and are performed only to demonstrate that the consequences of these events are bounded by the maximum hypothetical accident (MHA). With this licensing action, the CRE becomes the MHA, replacing the large break LOCA currently in the design basis. The control room habitability report (CRHR) is a living document, which is referenced in the Updated FSAR. As such, any changes to the plant design or procedures that affect assumptions or evaluations in the CRHR are subject to the provisions of 10 CFR 50.59. The NRC agrees with FPC's position that the assumptions and methods used in the control room habitability analyses for the SGTR and LLA need not be considered part of the design basis. The NRC based this decision on the fact that these analyses were performed only to identify the limiting accident with regard to GDC-19. The NRC's conclusion should not be interpreted as relaxing FPC's commitment to GDC-19, which requires consideration of all accident conditions evaluated as part of the CR-3 licensing basis.

The FPC application states that some analysis assumptions were based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and Draft RG 1081(DG-1081), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." NUREG-1465 is a technical report written primarily for advanced reactor designs as a basis for possible changes to regulatory requirements. The information in NUREG-1465 is limited to fuel with burnups less than 40 giga-watt days/ metric ton uranium (GWD/MTU). The NRC developed DG-1081 to establish guidance in using ASTs incorporating and expanding upon those aspects of NUREG-1465 deemed acceptable for operating reactors with burnups to 62 GWD/MTU. DG-1081 was published for public comment in December 1999. Because of public comments, revisions were made in the final RG. RG 1.183, issued for use in July 2000, supersedes DG-1081. The NRC cannot accept referencing NUREG-1465 as a part of the licensing basis because the document contains information that the NRC has not found to be acceptable as regulatory guidance for operating reactors. The NRC asked FPC to provide a statement of their intent to use the applicable guidance of RG 1.183, or accepted alternatives, in future AST applications at CR-3. FPC provided the requested license commitment in their letter of June 14, 2001.

### 3.2.1 Loss-of Coolant Accident (LOCA) Evaluation

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment and to confirm that the bases of the original plant site acceptance are not affected by the change. FPC assumes an abrupt failure of a large reactor coolant pipe and assumes that substantial core damage occurs due to this event. The assumption of core damage is conservative because DBA thermo-hydraulic analyses in the CR-3 Updated FSAR demonstrate the fuel damage thresholds are not exceeded.

Source Term Consideration

Fission products from the damaged fuel are released into the reactor coolant system (RCS) and then into the primary containment (CNMT). With the LOCA, it is anticipated that the initial fission product release to the CNMT will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins 30 seconds after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.3 hours. Tables 2, 4, and 5 of RG 1.183 define the source term used for these two phases. These data are summarized in Table 3-1.

Table 3-1 Release Fractions as a Function of Release Period

| Radionuclide Group | Gap Release *(0.5 hrs)* | Early In-Vessel *(1.3 hrs)* |
|---|---|---|
| Noble Gases (Xe, Kr) | 0.05 | 0.95 |
| Halogens (I, Br) | 0.05 | 0.35 |
| Alkaline Metals (Cs, Rb) | 0.05 | 0.25 |
| Tellurium Group (Te, Sb, Se) | 0 | 0.05 |
| Ba, Sr | 0 | 0.02 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0.0025 |
| Cerium Group (Ce, Pu, Np) | 0 | 0.0005 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0.0002 |

The inventory in each release phase is released at a constant rate over the duration of the phase, starting at the onset of the phase. Once dispersed in the CNMT, the release to the environment is assumed to occur through three pathways:

- Leakage of CNMT atmosphere (i.e., design leakage).
- Leakage from emergency core cooling systems (ECCS) that recirculate CNMT sump water outside of the CNMT (i.e., design leakage).
- Hydrogen purge release at 14 days into the accident.

For the control room, the dose contribution from CNMT shine, plume shine, and CREVs filter shine was considered.

The AST assumes that the iodine released to the CNMT is comprised of 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption of this iodine speciation is predicated on maintaining the CNMT sump water at pH 7.0 or higher. CR-3 provides for pH control with the dissolution of trisodium phosphate in the post-accident CNMT sump water. FPC has determined that the sump pH would range from 7 to 7.6. The NRC reviewed FPC's pH determination in its review of License Amendment No. 145 and found it to be acceptable.

Primary Containment (CNMT) Leakage Pathway

The CR-3 CNMT is projected to leak at its design leakage of 0.25 percent of its contents by weight per day for the first 24 hours and then at 0.125 percent for the remainder of the 30-day accident duration. FPC uses a two-region CNMT transport model in assessing the CNMT leakage pathway. This model is comprised of a region that is sprayed by the CNMT spray system and an unsprayed region. The sprayed region envelops 65.2 percent of the total free volume of the CNMT. The air in the two regions mixes at a rate equal to two turnovers of the unsprayed region volume per hour, and leaks to the environment at the design leakage rate. Radioactive decay is credited. FPC did not credit natural deposition as a removal process. FPC assumes that the sprays become effective at 124 seconds with an aerosol removal coefficient of 1.98 $hr^{-1}$ and an elemental removal coefficient of 19.81 $hr^{-1}$. FPC limits the elemental iodine decontamination factor (DF) to a maximum value of 200, which was determined to occur at 3.3 hours. No spray removal of elemental iodine is credited after that. Similarly, FPC decreases the aerosol removal coefficient by a factor of ten when the aerosol DF reaches 50, which was determined to occur at 4.8 hours. The removal coefficients are comparable to those previously used by FPC in earlier CR-3 analyses. The release of fission products from the CNMT to the environment occurs as an unfiltered ground level release. The CNMT leakage pathway modeling is consistent with the guidance in RG 1.183 and Standard Review Plan (SRP) Section 6.5.2.

Leakage from Emergency Core Cooling Systems

During the progression of a LOCA, some fission products released from the fuel will be carried to the CNMT sump via spillage from the RCS and by natural processes such as deposition and plate-out. Post-LOCA, the CNMT sump is a source of water for ECCS. Since portions of these systems are located outside of the CNMT, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ECCS, FPC assumes that all of the radioiodines released from the fuel are instantaneously moved to the CNMT sump. Noble gases are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides are not expected to readily become airborne on release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative in that all of the radioiodine released from the fuel is credited in both the CNMT atmosphere leakage and the ECCS leakage. In a mechanistic treatment, radioiodines in the CNMT atmosphere would relocate to the suppression pool over time.

The analysis considers the equivalent of 0.04 gpm ECCS leakage starting at the onset of the LOCA. This leakage includes a factor-of-two multiplier to address increases in the leakage due to normal material degradation between surveillance tests. FPC assumes that 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine species. The leakage enters the environment as an unfiltered ground level release.

Hydrogen Purge

Following a LOCA, hydrogen is expected to be generated in the CNMT because of various chemical reactions with metals in the reactor, reactor building, and because of radiolytic decomposition of water. Although there are provisions in the CR-3 design for the use of a hydrogen recombiner, CR-3 relies on controlled post-accident purging of the CNMT as a means of controlling hydrogen accumulation. This release pathway is modeled as described for the CNMT leakage pathway, with the exception of an increase in the rate of environmental release. For this assessment, it was assumed that the CNMT leak-rate would be 100 cfm starting at 14 hours post-accident and continuing for the remainder of the accident. These assumptions are conservative in that Updated FSAR, Appendix 14.B, projects that the hydrogen concentration would reach the 3.5 percent control level at 14.8 hours and that a purge rate of 50 cfm would be necessary to maintain hydrogen concentration acceptable. This event is a filtered, ground level release.

Offsite Doses

FPC evaluated the maximum 2-hour TEDE to an individual located at the exclusion area boundary (EAB) and the 30-day TEDE to an individual at the outer boundary of the low populations zone (LPZ). The resulting doses are less than the 10 CFR 50.67 criteria.

Control Room Doses

FPC evaluated the dose to operators in the control room. The CR-3 CCHE can be characterized as zone isolation with recirculation filtration. There is no intentional emergency pressurization. The recirculation filter system is manually aligned and made operational at about 30 minutes post-accident. The 37,800 cfm flow rate is assumed to flow through filters that are 95 percent efficient for iodine aerosols and 90 percent efficient for elemental and organic iodine chemical forms.

The normal air makeup to the control room prior to the activating event is 5700 cfm. This makeup air is terminated when the control room isolation occurs on receipt of either a high CNMT pressure, or loss-of-offsite power (LOOP) signal, or by manual operator action. For the LOCA and CRE (CNMT leakage case) accidents, the high CNMT pressure signal isolates the control room well before a released plume would reach the control room intakes. For the SGTR, MSLB, CRE (secondary leakage case) and LLA w/LOOP, FPC assumes that the LOOP that occurs at the time of reactor trip will result in control room isolation. For the FHA and LLA w/o LOOP, no isolation of the control room is credited. For the accidents involving releases via the secondary system (SGTR, MSLB, and CRE), LOOP is the limiting case since credit cannot be taken for iodine mitigation (1/10,000) in the main condenser if there is a LOOP.

FPC had previously performed integrated tracer gas testing of the CR-3 control room unfiltered inleakage and determined values for unfiltered inleakage less than 500 cfm. For the control room analyses for this amendment request, FPC has assumed an unfiltered inleakage of 1000 cfm. FPC proposed eliminating the requirement for periodic integrated testing based, in part, on the margin between the measured leakage of less than 500 cfm and the assumed analysis leakage of 1000 cfm. This unfiltered infiltration is assumed to commence at the onset of the event and continue for 30 days. The assumed 1000 cfm inleakage includes the 10 cfm access door leakage identified in SRP-6.4.

FPC had originally proposed using the margin between the 10 CFR 50.67(b)(2)(iii) dose criterion and the LOCA analysis control room dose result as a control room envelope breach margin in the CR-3 TSs. However, re-analysis of the CRE, in response to NRC requests for additional information, showed that the CRE event would be more limiting. Thus, the breech margin is now based on the CRE margin and is discussed in Paragraph 3.2.5 below.

The NRC is currently developing regulatory guidance regarding control room habitability, including surveillance testing of unfiltered inleakage. In addition, the Nuclear Energy Institute (NEI) has issued an industry initiative document (NEI-99-03) on control room habitability. The NRC's acceptance of an FPC unfiltered inleakage assumption here does not foreclose any future generic regulatory actions that may become applicable to CR-3 in this regard. FPC provided an unsolicited regulatory commitment in their June 14, 2001, letter to apply NRC-approved guidance for habitability envelope integrity monitoring and verification at CR-3, including proposal of ITS changes if appropriate.

Conclusion-LOCA

The NRC reviewed the analysis description and performed an independent calculation to confirm the FPC results. Based on this review, the NRC concluded that the licensee's application of the AST to the CR-3 LOCA analysis is acceptable. Table 3-2 provides the analysis assumptions found acceptable by the NRC. Table 3-3 provides the doses projected by FPC.

3.2.2 Main Steam Line Break (MSLB) Evaluation

The accident considered is the complete severance of a main steam line outside the CNMT but upstream of the MSIVs. The radiological consequences of a break outside CNMT will bound the results from a break inside CNMT. Because of this MSLB, the secondary water in the affected once-through steam generator (OTSG) completely flashes to steam. It is assumed that there is a primary-to-secondary leak that allows reactor coolant to leak into the affected OTSG at a rate of 1 gpm. Although this event does not result in fuel damage, the reactor coolant activity is based on steady state operation with 1.0 percent degraded fuel clad. No iodine spike is considered, as this is outside of the CR-3 design basis. For this particular analysis, the only required change from the original analysis was to replace the whole body and thyroid dose quantities with the TEDE quantity specified by 10 CFR 50.67. FPC made this correction via spreadsheets. The NRC reviewed these spreadsheets and found the adjustment to be acceptable.

FPC did not consider the control room doses for this event, based on its determination that the results of a MSLB would be bounded by those for a SGTR due to the significantly greater primary-to-secondary leakage in the SGTR event. FPC assumed no holdup or mitigation in the OTSGs for either the MSLB or SGTR. The NRC agrees with this conclusion in that the modeling of both events would involve similar analyses inputs with the exception of the primary-to-secondary leakage, which is greater for the SGTR.

Conclusion-LOCA

Based on this review, the NRC concluded that the licensee's application of the AST to the CR-3 MSLB analysis is acceptable. Table 3-2 provides the analysis assumptions found acceptable by the NRC. Table 3-3 provides the doses projected by FPC.

3.2.3 Steam Generator Tube Rupture (SGTR) Evaluation

The accident considered is the double-ended rupture of a single OTSG tube. The radiological consequences of this event are caused by the transfer of radioactive reactor coolant to the secondary side of the OTSG and the subsequent release of radioactive materials to the environment. FPC assumes that the primary-to-secondary break flow following a SGTR results in a depressurization of the RCS, a reactor trip, and actuation of safety injection. For the SGTR, FPC considered two separate scenarios: a licensing basis analysis for the offsite dose, and an SRP-like analysis for control room habitability.

For the licensing basis case, no single failure, no iodine spiking, and no LOOP are assumed. With no LOOP, the main condenser remains available to mitigate iodine releases to the environment. FPC assumes a primary-to-secondary release of 60.5 lb/sec for the entire 8 hour duration of the event. At 8 minutes into the event, a reactor trip occurs. Between 8 and 9 minutes, the release is via the atmospheric steam dump and the code safety valves. After 9 minutes, releases are via the main condenser. FPC assumes a main condenser iodine reduction factor of 1/10,000. The RCS activity is assumed to be that associated with 1 percent failed fuel. At 8 hours, the plant has been cooled sufficiently to use the decay heat system for further cool down. The NRC reviewed the analysis description and the associated spreadsheets and found them acceptable.

For the control room case, a single failure, LOOP, and iodine spiking are assumed. FPC assumes the initial iodine inventory in the RCS to be at the maximum concentration permitted by TSs. Two iodine-spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and that the RCS iodine inventory is at 60 μCi/gm dose equivalent I-131. The second case assumes the event initiates an iodine spike. Iodine is released from the fuel to the RCS at a rate 500 times the normal iodine appearance rate for a period of eight hours. FPC performed a thermo hydraulic analysis using RELAP5/MOD2-B&W code (BAW-10164PA) that estimates mass release and radioactivity releases for four time intervals following the event. In this scenario, the reactor trip at 8 minutes causes a LOOP resulting in control room isolation and loss of the main condenser. At 53 minutes, the atmospheric dump valve (ADV) on the unaffected OTSG fails to open, thereby delaying plant cool down and prolonging the release from the RCS. The failed ADV is opened at 78 minutes and the affected OTSG is isolated at 108 minutes, terminating its release. Plant cool down continues on the unaffected OTSG until 8 hours after the accident when all releases are terminated. The radioactivity releases were then input to the RADTRAD code and control room doses calculated.

Conclusion-SGTR

The NRC reviewed the analysis description and performed an independent calculation of the pre-accident spike case to confirm the FPC results. Based on this review, the NRC concluded that the licensee's application of the AST to the CR-3 SGTR analysis is acceptable. Table 3-2 provides the analysis assumptions found acceptable by the NRC. Table 3-3 provides the doses projected by FPC.

3.2.4 Fuel Handling Accident (FHA)

This accident analysis postulates that a spent fuel assembly is dropped during refueling, damaging all of the rods in the assembly. This accident could happen inside the CNMT or in the fuel handling building. The assumptions chosen for this evaluation are appropriate for either location. The core inventory is assumed to decay for 72 hours, the earliest that fuel movement will commence. The radial peaking factor of 1.8 was conservatively used to represent the most limiting fuel assembly. The entire gap inventory in the damaged rods is assumed to be released from the fuel. FPC assumed that 8 percent of the I-131 inventory of the core was in the fuel rod gap, along with 10 percent of the Kr-85 and 5 percent of all other iodines and noble gases. Alkali metals make a negligible contribution to dose for this analysis and are omitted. FPC assumed the iodine species fractions for the fuel release to be 95 percent aerosol, 4.85 percent elemental and 0.15 percent organic. This differs from the guidance in RG 1.183. However, since credit is not being taken for filtration in the auxiliary building or control room, the difference in the iodine species has no effect on the results and the assumption is, therefore, is acceptable. If filter credit were to be taken in a future application, the iodine species would need to be adjusted. FPC assumed an effective pool DF of 100. FPC assumed that the control room would not be isolated and that the CREVS recirculation filters would not be used. The normal air intake of 5700 cfm is assumed to continue for the 30-day accident duration.

Conclusion-FHA

The NRC reviewed the analysis description and performed an independent calculation to confirm the FPC results. Based on this review, the NRC concluded that the licensee's application of the AST to the CR-3 FHA analysis is acceptable. Table 1 provides the analysis assumptions found acceptable by the NRC. Table 2 provides the doses projected by FPC.

3.2.5 Control Rod Ejection Accident (CRE)

This accident analysis postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding is projected. This failure breeches the reactor pressure vessel head resulting in an LOCA to the CNMT. A reactor trip will occur. The release to the environment is assumed to occur through two pathways:

1. Release of CNMT atmosphere (i.e., design leakage)

2. Release of RCS inventory via primary-to-secondary leakage through the OTSGs

FPC assumed that 14 percent of the fuel rods suffer sufficient damage to result in the release of their entire gap inventory to the RCS and the CNMT. FPC assumed that 10 percent of the core inventory of noble gases and iodines are in the fuel rod gap. The thermo-hydraulic analyses show that fuel melt does not occur and there is no release from the fuel pellets. For the CNMT leakage case, all of the fission products released from the fuel are assumed to enter the CNMT. For the secondary release pathway, all of the fission products released from the fuel gap are assumed to remain in the RCS and be available for leakage to the secondary and the environment. A core radial peaking factor of 1.8 is conservatively assumed.

For the CNMT leakage case, the iodine released is 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. FPC assumes the iodine species in the secondary to be 97 percent elemental and 3 percent organic.

The CNMT is projected to leak at its design leakage of 0.25 percent of its contents by weight per day for the first 24 hours and then at 0.125 percent for the remainder of the 30-day accident duration. FPC does not credit reduction of iodine by CNMT sprays, but did credit natural deposition in the CNMT using the 10-percentile Powers model in the RADTRAD code.

A primary-to-secondary leakage of 150 gallon per day is assumed. No credit is assumed for holdup or iodine partitioning in the OTSG, as the primary-to-secondary is released immediately to the environment. FPC assumed that steaming would continue for
72 hours at which time the decay heat system would be placed in service.

For the CCHE dose analysis, FPC assumed that a LOOP would occur at T=0. This is conservative, as for LOOPs occurring later or not at all, mitigation in the main condenser would be available. Although the NRC does not generally credit such mitigation, credit for the main condenser mitigation has been previously taken in the CR-3 licensing basis.

FPC proposed using the margin between the 10 CFR 50.67(b)(2)(iii) dose criterion and the control room dose result for the secondary release case as a breach margin in the CR-3 TSs. FPC determined that the dose margin was equivalent to 400 cfm additional inleakage. The NRC confirmed the acceptability of an increase in the unfiltered inleakage from 1000 cfm to 1400 cfm by performing independent analyses at the higher flow rate.

Conclusion-CRE

The NRC reviewed the analysis description and performed an independent calculation to confirm the FPC results. Based on this review, the NRC concluded that FPC's application of the AST to the CR-3 CRE analysis is acceptable. Table 3-2 provides the analysis assumptions found acceptable by the NRC. Table 3-3 provides the doses projected by FPC.

3.2.6 Letdown Line Rupture Accident (LLA)

This accident analysis postulates a failure of a small line containing reactor coolant outside of the CNMT. The CR-3 licensing basis event considers a failure of a letdown line inside the auxiliary building. FPC assumes that a reactor trip occurs at 6 minutes following the rupture and that it takes 19.5 minutes from the start of the event to identify and isolate the letdown line break. Four break flow rates, based on thermo-hydraulic evaluations, were combined with reactor coolant specific activity to provide release rates for four time intervals. FPC assumes the initial iodine inventory in the RCS to be at the maximum concentrations permitted by TSs. Two iodine-spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and the RCS iodine inventory is at 60 μCi/gm dose equivalent I-131. The second case assumes the event initiates an iodine spike. Iodine is released from the fuel to the RCS at a rate 500 times the normal iodine appearance rate for a period of 8 hours. In both cases, 100 percent of the noble gases and 10 percent of the halogens in the leakage are assumed to enter the auxiliary building atmosphere. No credit was taken for iodine removal by the auxiliary building filters. These assumptions were applied in determining both offsite and control room doses.

FPC considered two cases, one with LOOP and one without LOOP. In both cases, the reactor trip occurs at 6 minutes into the event. For the LOOP case, the control room is isolated at the time of the LOOP due to loss of power. In this case, flow through the recirculation filters is manually established at 36 minutes after the event. An unfiltered inleakage of 1000 cfm is assumed. For the case without LOOP, FPC assumed that CREVS would remain in the normal alignment with an intake flow rate of 5700 cfm for the entire 30 days. The NRC reviewed the analysis description and performed an independent calculation of the pre-incident spike cases to confirm the FPC results.

Conclusion-LLA

Based on this review, the NRC concluded that FPC's application of the AST to the CR-3 LLA analysis is acceptable. Table 3.2 provides the analysis assumptions found acceptable by the NRC. Table 3-3 provides the doses projected by FPC.

3.2.7 Waste Gas Decay Tank Rupture (WGDT)

FPC evaluated the consequences of a release of the contents of all three waste gas decay tanks. The inventory in each tank is assumed to be the maximum allowable quantity specified in the offsite dose calculation manual (ODCM). ODCM

Specification 2.17, “Waste Gas Decay Tanks,” establishes a maximum allowable inventory for each tank. The associated bases (3.15) states that this restriction ensures that the dose at the EAB will not exceed 0.5 rem in the event that the contents of all tanks are simultaneously released in an uncontrolled manner.

Conclusion-WGDT

Since the analysis submitted for this amendment request is consistent with those bases and the analyzed results are less than 0.5 rem, there is reasonable assurance that the 5 rem TEDE criterion of 10 CFR 50.67 will be met. The NRC has determined this to be acceptable.

3.2.8 Other Radiological Consequence Analyses

RG 1.183 requires an applicant for an AST to review, and re-analyze as necessary, all design basis radiological consequence analyses for which one of more assumptions are affected by the proposed changes. As discussed above, FPC has re-analyzed the design basis accident analyses for the AST. FPC considered the impact of the AST proposed in this amendment in several additional CR-3 radiological consequence analyses. A study performed by the NRC determined that the results of dose analyses that assumed the Total Integrated Dose (TID) 14844 source term and that assessed the traditional whole body and thyroid dose quantities would generally bound TEDE analyses performed with an AST. The exception to this conclusion involves assessments for which the radiation source is contaminated CNMT sump water, either inside the CNMT or in ECCS systems outside of CNMT. The NRC study showed that analysis results obtained with the TID14844 source term and traditional dose quantities would bound the TEDE results using the AST, with all other analysis parameters held the same, for approximately the first 30 days and longer following the accident.

FPC considered the impact of the AST on current post-accident vital mission doses based on the TID14844 source term. FPC noted that the CR-3 vital missions occur during the first day post-accident with the exception of the CNMT purge that occurs 10-14 days after the accident. Since the period of performance is less than 30 days, FPC did not re-assess the mission doses for the AST. The NRC finds this position acceptable in that it complies with the guidance of RG 1.183.

Similarly, none of the assumptions used in the assessment of technical support center (TSC) habitability were affected by the changes to the control room isolation, leak rate, and monitor isolation actuation. As such, FPC did not re-assess the TSC doses for the AST. The NRC finds this position acceptable in that it complies with the guidance of RG 1.183.

FPC considered the impact of the AST on the design and range requirements for post-accident radiation monitoring and the post-accident sample system (PASS) doses. FPC noted that the design envelope for the radiation monitoring and the PASS includes worst-case DBA concentrations that bound those from an AST. As such, FPC concluded that there would be no adverse impact. The NRC finds this position acceptable in that it complies with the guidance of RG 1.183.

The total integrated dose (TID) for equipment qualification (EQ) could be affected by the increased cesium in the CNMT sump water if the EQ survivability period is greater than about 30 days. The survival period for some CR-3 EQ components is 6 months. FPC identified that its analyses show that the six-month TID could be 10-20 percent greater with the AST than with the TID14844 source term. It is FPC's position that this slight increase is compensated by significant conservatism included in the EQ TID assessments. FPC EQ analyses assign the worst dose in a plant zone to every component in that zone. FPC also assumes that every pipe run that could contain accident fluids does contain the design basis activity. FPC's analyses do not credit the shielding effects of intervening equipment. There is additional margin between the calculated TID and the TID acceptance criteria and the actual equipment failure point. The NRC notes that the impact identified by FPC is consistent with similar analyses performed by other licensees, and those by the NRC. While FPC's statements regarding analysis conservatisms are generally valid, the NRC based its acceptance of FPC's evaluation on regulatory guidance. RG 1.183 contains guidance that states that re-analysis of EQ TID calculations is warranted only if a plant modification associated with the AST implementation has affected one or more assumptions in existing EQ TID calculations. The guide provides that, pending the outcome of a generic safety issue review being performed by the NRC, the necessary re-analysis may be performed using the existing TID14844 source term. FPC has proposed no plant modifications that could impact EQ TID and, therefore, no re-analysis is necessary. The NRC has determined this to be acceptable.

### 3.2.9 Atmospheric Dispersion ($\chi$/Q)

FPC utilized $\chi$/Q values documented in the CR-3 Updated FSAR in the performance of the DBA analyses. No changes were made in these values. FPC assumes a single set of control room $\chi$/Q values for all accidents. In response to NRC questions, FPC stated that this set of $\chi$/Q values were originally developed for the shortest distance between the CNMT surface and the CCHE. These values were based on the wind speed and building wake correction; the lateral and vertical dispersion factors were not included. FPC further stated that the other release points considered in these assessments are located at greater distances away, but within the same wind direction sectors. These values were approved by the NRC as part of the NUREG-0737 Item III.D.3.4 review in 1989.

## 3.3 Proposed TS Changes

3.3.a. FPC has proposed a change to ITS 3.7.12, "Control Room Emergency Ventilation System (CREVS)," to remove applicability of the ITS during movement of irradiated fuel assemblies.

This change impacts only the FHA. FPC did not credit control room isolation or the use of the recirculation filters in the performance of the FHA analyses, as described in Paragraph 2.2.4 above. Since the dose criterion of 10 CFR 50.67(b)(2)(iii) and that of GDC-19 will be met without this credit, this proposed change is acceptable from a radiological consequences standpoint.

3.3.b FPC has proposed a change to ITS 3.7.12, LCOs such that an inoperable CCHE due to breach in excess of limit will require restoration within 24 hours (reduced from 7 days).

FPC's analyses demonstrate that the unfiltered inleakage could increase by 400 cfm (to a total of 1400 cfm) and the control room doses still meet the dose criterion of 10 CFR 50.67(b)(2)(iii) and that of GDC-19. As such, this proposed change is acceptable from a radiological consequences standpoint to the extent that a proposed breach will not result in unfiltered inleakage in excess of 400 cfm, which, when added to the nominal inleakage of 1000 cfm, results in a total unfiltered inleakage no greater than 1400 cfm.

3.3.c. FPC has proposed changes to ITS 3.7.12 LCO conditions such that an inoperable CREVS train (for reasons other than CCHE breach) will require restoration within 7 days (exempted CCHE breach) and that two inoperable CREVS trains (for reasons other than CCHE breach) will require plant shutdown.

These proposed changes do not impact the CCHE dose analyses. As such, these proposed changes are acceptable from a radiological consequences standpoint.

3.3.d. FPC has proposed revising SR 3.7.12.4 for performance of a periodic integrated leakage test. The current SR 3.7.12.4 expires at the end of cycle 12. The proposed revision maintains current design, operational, and maintenance administrative controls along with a commitment to adopt the NRC generic guidance on maintaining control complex habitability envelopes when it is issued.

FPC has provided evidence that two previous recent integrated tests have shown unfiltered inleakage rates less that 500 cfm. FPC performed the radiological analyses in support of the AST implementation assuming an unfiltered inleakage rate of 1000 cfm, twice that of the observed test results. The NRC finds the revised surveillance to be consistent with the intent of the guidance in NUREG-1430, and finds finds FPC's proposal acceptable.

3.3.e. FPC has proposed a change to ITS 5.6.2.12, "Ventilation Filter Testing Program," to remove the ABVES from the program.

FPC did not assume credit for the operation of the auxiliary building ventilation exhaust system in any of the radiological analyses submitted in support of the AST application and showed that the control room doses still meet the dose criterion of 10 CFR 50.67(b)(2)(iii) and that of GDC-19. As such, this proposed change is acceptable from a radiological consequences standpoint.

3.3.f. FPC has proposed a change to ITS 5.6.2.12, “Ventilation Filter Testing Program,” to change the methyl iodine penetration acceptance criteria for control room filters from “less than 2.5% “ to ”less than 5.0%.” (Filter efficiency reduced from 95 percent to 90 percent in the radiological analyses.)

FPC performed the CCHE control room dose analyses assuming 95 percent efficiency for aerosols and 90 percent efficiency for elemental and organic iodine species and showed that the control room doses still meet the dose criterion of 10 CFR 50.67(b)(2)(iii) and that of GDC-19. As such, this proposed change is acceptable from a radiological consequences standpoint.

3.3.g. FPC has proposed deletion of ITS 3.3.16, “Control Room Isolation - High Radiation.”

FPC performed the CCHE control room dose analyses assuming no credit for CCHE isolation actuated by these monitors and showed that the control room doses still meet the dose criterion of 10 CFR 50.67(b)(2)(iii) and that of GDC-19. As such, this proposed change is acceptable from a radiological consequences standpoint.

3.3.h. FPC has proposed changes to ITS 3.7.18, “Control Complex Cooling System,” to remove applicability of the ITS during movement of irradiated fuel assemblies.

FPC assumed that the control room would not be isolated during an FHA, and that the CREVS recirculation filters would not be used. They then showed the control room doses still meet the dose criterion of 10 CFR 50.67(b)(2)(iii) and that of GDC-19. Since this is the only factor to be considered, the NRC finds the proposed change acceptable.

3.3.i. FPC has proposed addition of a Note to T/S 3.7.12 that allows the CCHE boundary to be opened intermittently under administrative control.

This proposed change does not impact the CCHE dose analyses and is consistent with TSTF-287. As such, this proposed change is acceptable from a radiological consequences standpoint.

TABLE 3-2

RADIOLOGICAL ANALYSIS ASSUMPTIONS

Assumptions Common to One or More Analyses

| | |
|---|---|
| Reactor power, MWt (includes 1.02 factor) | 2619 |
| RCS specific activity, equilibrium μCi/gm dose equivalent I-131 | 1.0 |
| RCS specific activity, spike μCi/gm dose equivalent I-131 | 60.0 |
| Dose conversion factors | FGR11 and FGR12 |
| Control room volume, $ft^3$ | 365,000 |
| Normal ventilation makeup flow, cfm | 5700 |
| Control room filtered makeup flow, cfm | 0 |
| Control room recirculation filter efficiency percent | |
| Aerosol | 95 |
| Elemental | 90 |
| Organic | 90 |
| Control room recirculation flow rate, cfm | 37,800 |
| Manual actuation of control room recirculation, min | 30 |
| Control room unfiltered Inleakage, cfm | 1000 |
| Control room breathing rate, $m^3$/sec | 3.47E-4 |
| Control room occupancy factors | |
| 0-24 hours | 1.0 |
| 1-4 days | 0.6 |
| 4-30 days | 0.4 |

Control room c/Q, sec/$m^3$

| Period | |
|---|---|
| 0-8 hrs | 9.00E-4 |
| 8-24 hrs | 5.31E-4 |
| 1-4 days | 3.38E-4 |

Offsite c/Q, sec/$m^3$

| Period | EAB | LPZ |
|---|---|---|
| 0-2 hrs | 1.6E-4 | |
| 0-8 hrs | | 1.4E-5 |
| 8-24 hrs | | 1.5E-6 |
| 1-4 days | | 7.7E-7 |
| 4-30 days | | 4.5E-7 |

Offsite breathing rates, $m^3$/sec

| Period | |
|---|---|
| 0-8 hrs | 3.47E-4 |
| 8-24 hrs | 1.75E-4 |
| 1-30 days | 2.32E-4 |

## Assumptions for LOCA Analyses

Core Inventory — Calculated by RADTRAD

Core release fractions and timing-CNMT atmosphere

| Duration, hrs | 0.5000E+00 | 0.1500E+01 |
|---|---|---|
| Noble Gases: | 0.5000E-01 | 0.9500E+00 |
| Hologens: | 0.5000E-01 | 0.3500E+00 |
| Alkali met.: | 0.5000E-01 | 0.2500E+00 |
| Tellurium Group: | 0.0000E+00 | 0.5000E-01 |
| Strontium: | 0.0000E+00 | 0.2000E-01 |
| Barium: | 0.0000E+00 | 0.2000E-01 |
| Noble Metals: | 0.0000E+00 | 0.2500E-02 |
| Cerium Group: | 0.0000E+00 | 0.5000E-03 |
| Lanthanides: | 0.0000E+00 | 0.2000E-03 |

Core release fractions and timing-ECCS leakage

| Duration, hrs | 0.5000E+00 | 0.1300E+01 |
|---|---|---|
| Iodine: | 0.5000E-01 | 0.3500E+00 |
| Cesium: | 0.5000E-01 | 0.2500E+00 (not available for release) |

| Iodine species fraction | Atmosphere | Sump |
|---|---|---|
| Particulate/aerosol | 95 | 0 |
| Elemental | 4.85 | 97 |
| Organic | 0.15 | 3 |

| | |
|---|---|
| CNMT volume, $ft^3$ (fraction) | 2.0E6 |
| Sprayed | 1.304E6 (0.652) |
| Unsprayed | 6.96E5 (0.348) |
| CNMT region mixing | two turnovers of unsprayed volume per hour |
| CNMT release, %/day | 0.25 |
| After 24 hours | 0.125 |
| CNMT spray Actuation, sec | 124 |
| CNMT spray iodine removal coefficient, $hr^{-1}$ | |
| Aerosol | |
| Before DF=50 | 1.98 |
| After DF = 50 | 0.198 |
| Elemental | |
| Before DF=200 | 19.81 |
| After DF = 200 | 0 |
| Organic | 0 |

| | |
|---|---|
| Duration of release, days | 30 |
| CNMT sump volume, $ft^3$ | 45,370 |
| ECCS leak rate, gpm, (includes 2x multiplier) | 0.04 |
| Start of hydrogen purge, days | 14 |
| CNMT hydrogen purge flow rate, cfm | 100 |
| RB filter efficiency, all species, percent | 90 |
| Control Room Isolation, minutes | Instantaneous |

<u>Assumptions for MSLB Analyses</u>

| | |
|---|---|
| RCS activity, percent fuel failure | 1.0 |
| Primary-to-secondary release, gpm | 1.0 |
| Primary-to-secondary release duration, hours | 85 |
| Iodine partitioning | 0 |

<u>Assumptions for SGTR Analyses - Offsite Case</u>

| | |
|---|---|
| RCS activity, percent failed fuel | 1.0 |
| Primary-to-secondary break flow, lbm/sec | 60.5 |
| Break flow duration, hours | 8 |
| Fraction of release via main condenser | |
| 0 to 8 minutes | 1.0 |
| 8 to 9 minutes | 0.45 |
| 9 minutes to 8 hours | 1.0 |
| Iodine mitigation by main condenser factor | 10,000 |

<u>Assumptions for SGTR Analyses - Control Room Case</u>

| | |
|---|---|
| RCS activity, µCi/gm d.e.I-131 | |
| Equilibrium case | 1.0 |
| Pre-incident iodine spike case | 60.0 |
| Co-incident iodine spike multiplier | 500 |
| Event time sequence | |
| Rx trip, LOOP, CCHE isolation, min | 8 |
| ADV fails shut on affected SG, min | 53 |

| | |
|---|---|
| OTSG isolated, min | 108 |
| Release terminated, hours | 8 |
| Primary-to-secondary flow | |
| 0-0.13 hours, lbm/sec | 33 |
| 0.13-0.63 hours, lbm/sec | 36 |
| 0.63-1.8 hours, lbm/sec | 30 |
| 1.8-8 hours, gpd via unaffected OTSG | 150 |
| Holdup and/or removal in secondary, DF | |
| Prior to 0.13 hours | 100 |
| After 0.13 hours | 1 |
| Control Room Isolation | @LOOP |

<u>Assumptions for FHA Analysis</u>

| | |
|---|---|
| Number of fuel assemblies in core | 177 |
| Number of damaged assemblies | 1 |
| Radial peaking factor | 1.8 |
| Fuel rod gap fractions | |
| I-131 | 0.08 |
| Kr-85 | 0.10 |
| All others | 0.05 |
| Iodine species fractions (see text) | |
| Elemental | 0.0485 |
| Organic | 0.0015 |
| Particulates | 0.95 |
| Pool scrubbing factor, effective | 100 |
| Control Room Isolation | none |

<u>Assumptions for Rod Ejection Accident Analyses</u>

| | |
|---|---|
| Core Inventory | |
| RADTRAD | |
| Radial peaking factor 1. | 8 |
| Fraction of rods that exceed DNB | 0.14 |
| Gap fraction | |
| Noble gases | 0.1 |
| Iodines | 0.1 |
| Fraction of rods that exceed DNB that experience melt | 0.0 |

| Iodine species fraction | Atmosphere | SG |
|---|---|---|
| Particulate/aerosol | 95 | 0 |
| Elemental | 4.85 | 0.97 |
| Organic | 0.15 | 0.03 |

| | |
|---|---|
| Containment release, percent /day | |
| 0-24 hours | 0.25 |
| 24-720 hours | 0.125 |
| Duration of CNMT release, days | 30 |
| Secondary release source term | Fuel |
| Primary-to-secondary leakage, all OTSG, gpd | 300 |
| RCS cold liquid volume, gal | 61,000 |
| Secondary release duration, hrs | 72 |
| Steam partition coefficient | 0.0 |
| Control Room Isolation | @LOOP |

Assumptions for Letdown Line Accident

| | |
|---|---|
| RCS activity, µCi/gm d.e.I-131 | |
| Equilibrium case | 1.0 |
| Pre-incident iodine spike case | 60.0 |
| Co-incident iodine spike multiplier | 500 |
| Primary side release, lbm/sec | |
| 0-0.033 hours | 110 |
| 0.033-0.1 hours | 100 |
| 0.1-0.22 hours | 95 |
| 0.22-0.325 hours | 95 |
| Primary side release, lbm | 114,000 |
| Auxiliary building filtration | none |
| Control Room Isolation | @LOOP |

Assumptions for Waste Decay Tank Rupture

| | |
|---|---|
| Number of tanks released | 3 |
| Tank inventory | |
| ODCM | |
| Release duration | instantaneous |
| Auxiliary building filtration | none |
| Control Room Isolation | none |

TABLE 3-3

RADIOLOGICAL ANALYSIS RESULTS[1], REM TEDE

| Event | 0-2 hr EAB | 30-day LPZ | 30-day CR |
|---|---|---|---|
| Main Steam Line Break | 0.005 | 0.003 | (2) |
| Steam Generator Tube Rupture | 0.14 | 0.046 | |
| Pre-incident Spike[3] | 5.98 | 0.523 | 1.19 |
| Co-incident Spike[3] | 2.40 | 0.21 | 0.365 |
| Fuel Handling Accident | 0.83 | 0.073 | 4.43(4) |
| Control Rod Ejection Accident | | | |
| CNMT | 1.03 | 0.288 | 0.754 |
| Secondary | 2.10 | 0.819 | 3.49 |
| Letdown Line Accident | | | |
| LOOP pre-incident | 0.614 | 0.054 | 0.895 |
| LOOP co-incident | 0.078 | 0.007 | 0.060 |
| No LOOP pre-incident | 0.614 | 0.054 | 3.24 |
| No LOOP co-incident | 0.078 | 0.007 | 0.339 |
| Loss-of-Coolant Accident | 7.59 | 1.107 | 2.30 |
| Waste Gas Decay Tank | 0.125 | 0.011 | (2) |

Notes

1. Determined by FPC and confirmed by NRC.
2. Not calculated. Deemed to be bounded by other analyzed accidents
3. Information case only (see text)
4. Although the FHA dose is greater, the CRE is the basis for the breach margin since the FHA did not assume any control room isolation.

3.4 Hazardous Chemical Analysis Evaluation

In CR-3 the presence of chlorine and sulfur dioxide pose potential liability to the habitability of the control room. These chemicals are used for water treatment and are stored at the Helper Cooling Tower site and at the site of the Unit 4 and Unit 5 cooling towers. These sites are located at 3400 ft and 3600 ft from the air inlet to the Unit 3 CCHE, respectively. Three tons of chlorine and 17 tons of sulfur dioxide are stored at the Helper Cooling Tower. The Crystal River Units 4 and 5 site has eight 1-ton cylinders of chlorine and one 1-ton cylinder of sulfur dioxide. FPC replaced the potentially significant sources of gaseous chlorine and sulfur dioxide in Units 4 and 5 with sodium hypochlorite and sodium bisulfite.

FPC evaluated habitability of the control room using the two minutes criterion, specified in Section 6.4 of the Standard Review Plan (SRP). This criterion requires that at least 2 minutes should elapse between detection of the presence of a toxic chemical in the control room and the time its concentration reaches toxic limits. The leakage of toxic chemical to the control room can be minimized by its timely isolation, which can be achieved either by operator action involving olfactory detection of toxic chemical and manual isolation, or by an automatic isolation on a signal from the toxic chemical monitors. The licensee has determined that the chlorine and sulfur dioxide 2-minute requirement is met for these chemicals stored on plant site without using automatic isolation mode.

The NRC evaluated the licensee's analyses and performed its independent verification using the NRC's computer code HABIT. It concludes that the assumptions made by the licensee and its methodology used in calculating the amounts of chlorine and sulfur dioxide leaking into the CCHE are in conformance with the requirement of Section 6.4 of the SRP. Also, the 2-minute criterion for the chlorine and sulfur dioxide stored on the plant site could be met without recourse to the automatic isolation of the control room on a signal from the chemical monitor. Relying on olfactory detection and manual isolation is adequate for maintaining habitability of the control room during an accidental release of these chemicals. However, for the release of chlorine during the transportation accident, habitability of the control room has to be protected by the automatic CCHE isolation activated by a signal from the chlorine monitor.

The NRC finds the FPC analysis to be acceptable.

### 3.5 Environmental Qualification Analysis Evaluation

The licensee states that the EQ doses have not been re-analyzed using an AST and current EQ doses based on Technical Information Document (TID)-14844 source term provide adequate justification for the continued operability of EQ components. The NRC agrees with this determination for the following reason:

The NRC has considered the potential impact of the postulated cesium concentration on the operability of safety systems at current operating reactors (*Federal Register*, 64 FR 72001). NRC analyses have shown that the EQ doses determined using the current TID-14844 source term are more limiting than those calculated using the NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," source term for exposure periods less than 30 days to 4 months following the accident. The postulated increase in the cesium concentration is not a concern for those systems and components having a safety function that is performed and completed earlier than 30 days following an accident. The NRC concludes that continued plant operation does not pose a threat to public health and safety since this equipment will remain capable of performing its intended design functions. The small increase in long-term dose is typically compensated by the significant conservatism incorporated in EQ dose assessments. This conservatism includes: application of the worst dose in a zone to most components in the zone, assumption that all piping that could contain post-accident activity doses contain such activity, and neglecting the shield effects of intervening equipment. In addition, significant margin typically exists between a component's calculated dose and the component's qualification dose. For most components, there is also additional margin between the qualification dose and the dose that would cause component failure. Generic Safety Issue (GSI)-187 will evaluate the validity of the postulated increase in the cesium concentration and determine whether further regulatory action is required. This resolution will apply to all operating plants and not just those making an AST submittal. In addition, in RG 1.183, the NRC has specified that a licensee making an AST submittal can continue to use the TID-14844 source term for EQ pending the resolution of the GSI.

The NRC approves the licensee's determination that current EQ doses based on TID-14844 source term provide adequate justification for the continued operability of EQ components.

### 3.6 Conformance to TSTF-287 Evaluation

LCO 3.7.12 has been modified by a Note allowing the CCHE boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other designed openings such as dampers and equipment or personnel access hatches, these controls consist of stationing at the opening a dedicated individual who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room area isolation is indicated.

If the control room boundary is inoperable in MODES 1, 2, 3, or 4 such that the CREVS trains cannot establish or maintain the required leakage rates, action must be taken to restore an OPERABLE control building boundary within 24 hours. The 24-hour Completion Time is reasonable based on the low probability of a design basis accident occurring during this time period and compensatory measures available to the operator to minimize the consequences of potential hazards.

The proposed changes would allow 24 hours (during Modes 1, 2, 3, or 4) to restore the capability to maintain control building boundary pressure before requiring the unit to perform an orderly shutdown and would also allow intermittent opening of the control room boundary under administrative control. Per the LCO Note, during the period that the control building boundary is inoperable, appropriate compensatory measures consistent with the intent of 10 CFR Part 50, Appendix A, GDC-19 will be utilized to protect the control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature, and relative humidity and to ensure physical security. These preplanned measures will be available to address these concerns for intentional and unintentional entry into the condition. For example, when the control boundary is breached by designed openings such as dampers and equipment or personnel access hatches, entry through doors, the proposed bases state, in addition to other necessary measures, that a dedicated individual will be stationed in the area in continuous contact with the control room to rapidly restore the boundary.

For inadvertent or preplanned breaches per maintenance or design modification in excess of design basis leakage limits, the proposed changes would allow 24 hours (during Modes 1, 2, 3, or 4) to restore the capability to maintain control building boundary pressure before requiring the unit to perform an orderly shutdown. It is required that procedures be available to provide compensatory actions for the control room operators and to have procedures in place to seal the breach in the unlikely event of an accident.

This proposed change is considered acceptable because of the low probability of an event requiring an intact control room boundary during the 24-hour ACTION Completion Time associated with Condition B.

Currently SR 3.7.12.4 requires an integrated leak test every 24 months to verify CCHE boundary integrity. This SR is only valid until completion of Fuel Cycle 12 at which time it expires. In order to implement TSTF-287, there is a premise that there is an SR that verifies the integrity of the control room boundary. This requirement has usually been met using some type of pressure test. Since the CR-3 envelope is a designed neutral pressure boundary, the standard SR written in TSTF-287 is not applicable. Thus, FPC has proposed a program to verify CCHE boundary integrity.

FPC has stated that CR-3 conforms to 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, for the CCHE and CREVS as follows:

- CREVS and CCHE are in the scope of the 10 CFR 50.65.
- CREVS is treated as risk significant.
- CCHE is monitored for system functional failures.
- CREVS is monitored for system functional failures and unavailability.
- CREVS is currently in the scope of the 10 CFR 50.65(a)(2), which exempts it from the monitoring specified in 10 CFR 50.65(a)(1).

FPC's preventive maintenance program for CREVS and CCHE include the following plant procedures:

- Doors - Surveillance and maintenance procedures: SP-805A - Fire Door Inspections, SP-805 - Fire Door Surveillance, and PM-175 - Fire Door Maintenance.
- CCHE Penetrations/seals - Surveillance and maintenance procedures: SP-407 - Fire Barrier Seals and MP-805 - Penetration Seals.
- Dampers - Surveillance and maintenance procedures: CS5026 - CREVS Damper inspection, PM-139 - HVAC Equipment Checks, and SP-353 - Monthly Functional Test.
- Floor Drains -maintenance procedure: CS-5295 - Loop Seal maintenance controls.

- The CCHE Breach Control procedure: Breaches are opened under administrative controls in accordance with ITS. Additionally, FPC has stated that the configuration control of the CCHE is maintained utilizing the plant documents as follows:
  - 3.1 - TDBD 9.4 - Control Room Habitability Report establishes design basis
  - 3.2 - NEP- 210A - Modification checklist addresses CCHE

FPC has also stated that the function failure of the system, structure, or component is identified, evaluated, and appropriate corrective action is taken by utilizing above plant procedures to preclude any condition where it cannot meet its intended Maintenance Rule function.

The NRC conducted an audit-based review of the preventive maintenance and surveillance procedures (i.e., MP-508 and SP-407), to evaluate the TS 5.6.2.21, Control Complex Habitability Envelope Integrity Program. The NRC finds that FPC's proposed CCHE integrity program is consistent with the proposed SR 3.7.12.4 to maintain the CCHE, if it is breached. Since the CREVS and CCHE are currently in the scope of the 10 CFR 50.65 at CR-3, and the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate plant preventive maintenance and surveillance procedures to maintain its intended function, the NRC finds the proposed CCHE integrity program an acceptable surveillance.

Since the basis of the amendment request is now TSTF-287, FPC's responses to Request for Information Questions 7 and 8 and Attachment E (Design Calculation Number M97-0137) in FPC's letter to the NRC dated June 14, 2001, are no longer applicable. In telephone correspondence between the NRC and FPC on August 22, 2001, and as stated in their letter dated August 28, 2001, FPC committed to remove CR-3 Design Calculation Number M97-0137, Revision 4 as an input to the Control Room Habitability Report and from the CR-3 breach margin control program. In the August 28, 2001, letter FPC removed Appendix A "Control Room Habitability Report" from their October 3, 2000, and June 14, 2001, letters. The FPC design basis documents provide design limits, as well as defined acceptable leakage margins, based on prior empirical testing and engineering analysis. The FPC CCHE Control Program requires that breaches in the habitability envelope are managed and evaluated to ensure that in-leakage remains below analysis limits. The FPC program requires an engineering evaluation to verify that the leakage margin as specified in the design basis documents is not exceeded. The NRC finds this acceptable.

FPC has committed to maintaining the current design control, operations procedures and maintenance procedures as the basis for the Control Complex Habitability Program. Additionally FPC has committed to accepting the NRC guidance to be issued in the near future as a basis for enhancing the CR-3 CCHE surveillance program described in ITS 5.6.2.21. These surveillance requirements are consistent with NUREG-1430, meet the intent of TSTF-287, and the NRC finds them to be acceptable.

Based on the low probability of an event occurring in this time and the availability of compensatory measures consistent with GDC-19 to minimize the consequences during an event, the proposed change is considered acceptable and the changes meet the intent of TSTF-287.

3.7 REGULATORY COMMITMENTS

FPC has provided the following regulatory commitments to support the amendment request.

| Commitment | Due Date |
|---|---|
| FPC will continue to apply current administrative controls for the identifying, tracking, and closing CCHE breaches, and continue to perform required inspection and maintenance activities for CREVS dampers, CCHE doors, penetration seals, and floor drains, to ensure margin is maintained between actual in-leakage and the 1000 cfm value assumed in the dose calculations. | Upon issuance of this requested License Amendment |
| Recognizing the importance of defense in depth afforded by the automatic control room isolation on high radiation, CR-3 will continue to test and maintain this feature. | Upon issuance of this requested License Amendment |
| CR-3 commits to using applicable assumptions of Regulatory Guide 1.183, Rev. 0 or acceptable alternatives thereto in future revisions to the design basis public and control room dose assessments. | Following NRC approval of LAR 262. |
| FPC commits to apply NRC approved guidance for habitability envelope integrity monitoring and verification at CR-3, including proposal of ITS changes if appropriate. | Following NRC approval of guidance on habitability envelope integrity monitoring and verification. |

The NRC has reviewed these commitments and finds them to be acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC has 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 69060). Accordingly, the 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 the amendment.

## 6.0 CONCLUSION

The NRC reviewed the assumptions, inputs, and methods used by FPC to assess the radiological impacts of the proposed changes. In performing this review, the NRC relied upon information placed on the docket by FPC, NRC experience in doing similar reviews and, where deemed necessary, on NRC confirmatory calculations. The NRC finds that FPC used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, the proposed TS changes, and the future power uprate. The NRC compared the doses estimated by FPC to the applicable criteria and to the results of confirmatory analyses by the NRC. The NRC finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room total effective dose equivalent due to postulated DBAs at CR-3 comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. The NRC finds, with reasonable assurance, that CR-3 AST implementation will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the CR-3 design basis is superseded by the AST proposed by FPC in its application of October 3, 2000. The previous offsite, control room, and TSC accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements should address all characteristics of the AST and the TEDE criteria as described in the now-updated CR-3 design basis. These changes required re-evaluations by FPC of certain analyses in the Control Room Habitability Report, which included the analyses for protection of the control room operators in the event of a toxic chemical release accident. To verify the analyses supporting FPC's amendment request, the NRC did not need to review the Control Room Habitability Report, as other supporting documentation was provided. The CR-3 Control Room Habitability Report is an FPC controlled document and will be maintained and updated per the FPC design control program. The re-evaluation performed by the licensee has indicated that the proposed amendments to the ITS will not change the original conclusions that the existing procedures will ensure protection of the control room operators during an accidental release of chlorine or sulfur dioxide in the Crystal River plant.

The NRC has additionally reviewed FPC's letter dated January 14, 1998, which provided specific commitments made by FPC concerning a JCO provided to the NRC in FPC's letter dated December 15, 1997. The January 14, 1998, letter referred to FPC submitting an updated and revised Control Room Habitability Report to provide additional technical information to support the JCO. The NRC has determined that the information provided by FPC in its submittal dated October 3, 2000, and revised August 28, 2001, has acceptably provided the information without the submittal of a revised Control Room Habitability Report. With the issuance of this amendment the NRC finds that FPC has met all commitments stated in the January 14, 1998, letter.

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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