Florida Power
A Progress Energy Company

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

August 28, 2001
3F0801-03

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River - Unit 3 – License Amendment Request #262, Revision 2, "Alternative Source Term and Control Room Emergency Ventilation System" (TAC No. MB0241)

References: 1. FPC to NRC letter, 3F0601-03, dated June 14, 2001, License Amendment Request #262, Revision 1, and Response to NRC Request for Additional Information RE: License Amendment Request #262, Revision 0, "Alternative Source Term and Control Room Emergency Ventilation System" (TAC No. MB0241)

2. FPC to NRC letter, 3F1000-08, dated October 3, 2000, License Amendment Request #262, Revision 0, "Alternative Source Term and Control Room Emergency Ventilation System"

Dear Sir:

Florida Power Corporation (FPC) hereby submits License Amendment Request (LAR) #262, Revision 2, which modifies the requests made in References 1 and 2. This request incorporates the proposed resolution of NRC staff comments from several conference calls that occurred following our response in Reference 1.

The proposed revision adopts Technical Specification Task Force Traveler #287 (TSTF-287) as the basis of the request. Revisions are being made to Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) Surveillance Requirement 3.7.12.4, the Bases for ITS Section 3.7.12 and a new Section 5.6.2.21, Control Complex Habitability Envelope Integrity Program, which is added to support the adoption of TSTF-287. Since the basis of LAR #262 is now TSTF-287, the response to RAI Questions 7 and 8 and Attachment E in Reference 1 are no longer applicable. FPC also withdraws Appendix A to References 1 and 2, CR-3 Control Room Habitability Report. Portions of the previous submittals regarding toxic gas are retained as an appendix to this letter. Additionally, LAR #262 Revision 1, Appendix B has been revised to be consistent with this submittal.

In order to facilitate review of the proposed changes, in addition to the ITS pages revised for this submittal, a consolidation of all three sets of revised ITS pages are attached. Please note that since References 1 and 2 were submitted, several changes were made to the Background Section of Bases 3.7.12, under the Technical Specifications Bases Control Program.

A001

FPC requests NRC approval of LAR #262 by September 21, 2001. The requested approval date will allow sufficient time to support activities in Refueling Outage 12, scheduled to begin on September 29, 2001. FPC will implement the proposed LAR within 45 days of issuance.

CR-3 has determined that this revised request does not change the conclusion in the Environmental Impact Evaluation and does not change the conclusions reached in the No Significant Hazards Consideration Determination submitted in Reference 2.

The CR-3 Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

As stated in Reference 2, the proposed ITS changes and the associated analysis provide assurance that 10 CFR 50 Appendix A General Design Criteria 19 limits are met and supersede commitments made in support of the interim Justification for Continued Operation. The changes requested in this letter do not impact the commitments made in Reference 1. This letter establishes no new regulatory commitments.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

John J. Holden
Director, Site Operations

JJH/pei

Attachments:

A. Description and Justification for Proposed Changes
B. Proposed Revised Improved Technical Specifications Change Pages – Strikeout/ Shadowed Format
C. Proposed Revised Improved Technical Specifications Change Pages – Revision Bar Format
D. Consolidation of LAR #262 Revisions 0, 1 and 2, Proposed Revised Improved Technical Specifications Change Pages – Revision Bar Format

Appendices:

A. Toxic Gas Report
B. Revised Summary of Radiological Analyses for FSAR Chapter 14 Accidents

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector

STATE OF FLORIDA

COUNTY OF CITRUS

John J. Holden states that he is the Director, Site Operations, Crystal River Nuclear Plant for Progress Energy; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

J J Holden

John J. Holden
Director, Site Operations

The foregoing document was acknowledged before me this 28th day of August, 2001, by John J. Holden.

Lisa A Morris

Signature of Notary Public
State of Florida

LISA A. MORRIS
Notary Public, State of Florida
My Comm. Exp. Oct. 25, 2003
Comm. No. CC 879691

LISA A. MORRIS

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known X -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT - 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #262, REVISION 2
Alternative Source Term and
Control Room Emergency Ventilation System

Description and Justification for Proposed Changes

Description of Proposed Changes

Florida Power Corporation (FPC) requests that the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) be modified as described below.

In License Amendment Request (LAR) #262, Revision 0, FPC proposed that Surveillance Requirement (SR) 3.7.12.4 be deleted. FPC requests that instead, a revised SR be inserted as follows:

Surveillance	Frequency
SR 3.7.12.4 Verify control complex habitability envelope integrity in accordance with 5.6.2.21.	In accordance with the Control Complex Habitability Envelope Integrity Program

The Control Complex Habitability Envelope Integrity Program is defined in the new Section 5.6.2.21.

5.6.2.21 Control Complex Habitability Envelope Integrity Program

A program shall be established to maintain the integrity of the control complex habitability envelope to ensure the dose limits of 10 CFR 50 Appendix A General Design Criteria 19 are not exceeded. The program shall establish acceptable leakage limits, ensure maintenance activities are monitored and provide a preventive maintenance program for the control complex habitability envelope.

The Control Complex Habitability Envelope Integrity Program shall ensure that:

1. Breaches in the habitability envelope are managed to ensure that in-leakage remains below design basis analysis limits.

2. The preventive maintenance program includes doors, wall/roof/floor penetrations, dampers and floor drains that are part of the control complex habitability envelope.

3. Periodic evaluations of the systems, components and key analysis assumptions are performed.

4. Configuration control of the CCHE is maintained.

The Bases for ITS 3.7.12 were also revised to delete references to the breach margin calculation, add references to the Control Complex Habitability Envelope Integrity Program and to make the Bases more similar to Technical Specification Task Force (TSTF) Traveler #287 (TSTF-287).

Justification for Proposed Changes

The proposed SR 3.7.12.4 is different than that included in NUREG-1430, Revison 2, "Babcock and Wilcox Standard Technical Specifications" due to the design of the CR-3 Control Complex Habitability Envelope (CCHE). The NUREG-1430 SR provides a requirement to verify the capability of maintaining a positive pressure. The CR-3 CCHE is a non-pressurized design and therefore, this SR is not appropriate. In its place, FPC requests that a SR be added to establish a CCHE Integrity Program to verify that CCHE integrity is maintained. The program will incorporate NRC guidance on CCHE integrity monitoring and verification following issuance of the final guidance. Until that time, the CCHE Integrity Program will be based on industry guidance and engineering judgment. The proposed SR will ensure that known breaches are controlled and maintained below the values determined in the design basis analysis. The CCHE Integrity Program will also include a preventive maintenance program to ensure that previously undetected breaches are identified and repaired in a timely manner. In addition, the program will include periodic evaluations of systems, components and key analysis assumptions and ensure configuration control of the CCHE.

CR-3 currently has programs and procedures in place that provide assurance that CCHE integrity is being properly maintained. Five key aspects of these programs and procedures are described below:

1. Maintenance Rule 10CFR50.65

    1.1 – CREVS and CCHE are in the scope of Maintenance Rule
    1.2 – System is treated as risk significant
    1.3 – CCHE and CREVS are monitored for system functional failures (FF)
    - A FF is any failure that identifies a condition where the system, structure or component (SSC) cannot meet its intended Maintenance Rule function.
    - For CREVS, the FFs include failure of a fan to start, run or provide design basis flows. Fan dampers failing to open would also constitute a FF.
    - For the CCHE, FFs include failure of a damper to close, failure of a damper to indicate closed or any boundary component identified to result in leakage greater than that assumed in the design basis analysis.

    1.4 – CREVS is monitored for system unavailability
    1.5 – Performance criteria include:
    - Unavailability less than 160 hours per 24 months for each CREVS train
    - Less than three total functional failures for CREVS and CCHE in 24 month period
    - No repeat failures in a 24 month period

    1.6 – System is currently a(2)

2. Quality Assurance

    2.1 – Safety-related system/structure included in scope of Appendix B
    2.2 – Quality control inspections required for CCHE seal repairs

2.3 – Nuclear Assessment Section assesses modification and maintenance processes including independent safety review of activities requiring full 10CFR50.59 evaluations
2.4 – Conditions adverse to quality tracked by Corrective Action Program

3. Configuration Control

3.1 – TDBD 9.4 –Establishes CCHE design basis
3.2 – NEP- 210A – Modification checklist addresses CCHE
3.3 – NEP-213 – Calculation review process addresses CCHE

4. Preventative Maintenance Program

4.1 – Doors- Surveilled and maintained by procedures:

- SP-805A – Includes annual fire door and seal inspections
- SP-805 – Includes monthly fire door and seal inspections
- PM-175 – Fire Door Maintenance

4.2 – CCHE Penetrations/seals – Surveilled and maintained by procedures:

- SP-407 – Fire Barrier Seal inspection – all seals inspected every 15 years, approximately ten percent every eighteen months
- MP-805 – Penetration Seals

4.3 – Dampers – Surveilled and maintained by procedures:

- CS5026 – CREVS Damper inspection – 24 month interval
- PM-139 – HVAC Equipment Checks
- SP-353 – Includes: Monthly functional test of CCHE isolation, Monthly run of CREVS trains, 24 month CREVS actuation on Engineered Safeguards signal

4.4 – Floor Drains –Maintained by:

- CS-5295 – Loop seal maintenance controls, monthly fill of loop seals

5. CCHE Breach Control

5.1 – Breaches maintained within design basis analysis, controlled by procedures:

- CP-147/CP-137

5.2 – Breaches opened under administrative controls in accordance with ITS

These controls were developed during the 1995 to 1997 time frame and have proven to be effective in maintaining the CCHE boundary. This conclusion is supported by the integrated leakage tests performed at CR-3 in 1997 and 1999. A significant effort was made in 1997 to minimize leakage into the CCHE. Following this effort, an integrated leakage test was performed in October 1997 with a resulting leakage rate of 523 cubic feet per minute (cfm). During the next two years, several modifications were done including the installation of a third emergency feedwater pump and upgrades to the high pressure and low pressure emergency core cooling systems systems. These modifications involved adding a significant number of cables and conduits that penetrate the CCHE. In addition, a number of CCHE doors and door seals were repaired or replaced. The penetrations and structures were restored using the plant processes and procedures described above. In September 1999, no preconditioning or sealing activities were performed prior to performing the integrated leak test. The resulting CCHE leakage was 513 cfm, less than the 523 cfm leakage found in 1997. Therefore, these tests confirm that the procedures and processes in place adequately restore the CCHE following maintenance and modifications.

For CR-3, maintaining the CCHE in-leakage below the values determined in the design basis analysis ensures that doses will remain under Title 10 Code of Federal Regulations Part 50 (10CFR50) Appendix A General Design Criteria 19 (GDC 19) limits. Therefore, although the proposed SR differs from NUREG-1430, the intent to ensure integrity of the CCHE is similar in purpose.

The proposed changes are also supported by the justification provided in TSTF-287.

A number of changes are being made to the Bases to support the adoption of TSTF-287. A paragraph is added to explain the Note to the ITS Limiting Condition for Operation 3.7.12. The Note allows normal ingress and egress to and from the CCHE. The Note also addresses administrative controls for planned openings such as opening hatches, panels and access ports of the CCHE. When implementing these preplanned administrative controls, Action Condition B need not be entered. The Bases for Action Condition B states that the CCHE boundary is inoperable when it is degraded such that the CREVS trains cannot perform their intended functions. Breaches of the envelope that are either planned or discovered may be evaluated in accordance with design basis documents to determine if the operability of the CCHE is compromised. Any actions taken to restore the CCHE boundary must be consistent with the intent of GDC 19. The Bases for SR 3.7.12.4 states that the Control Complex Habitability Envelope Integrity Program will ensure that CCHE integrity is maintained. Failure to meet individual requirements of the program does not necessarily make the CCHE inoperable. Each individual failure should be evaluated in accordance with design basis documents to determine if the CCHE can still perform its safety function. If the CCHE can still function as required in the design basis analysis, the system remains operable.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR - 72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #262, REVISION 2
Alternative Source Term and
Control Room Emergency Ventilation System

Proposed Revised Improved Technical Specifications and Bases Change Pages

Strikeout / Shadow Format

~~Strikeout Text~~	Indicates deleted text
Shadowed text	Indicates added text

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each CREVS train for ≥ 15 minutes.	31 days
SR 3.7.12.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.3	Verify each CREVS train actuates to the emergency recirculation mode on an actual or simulated actuation signal.	24 months
SR 3.7.12.4	Verify control complex habitability envelope integrity in accordance with ITS 5.6.2.21.	~~24 months~~In accordance with the Control Complex Habitability Envelope Integrity Program

~~NOTE~~

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

a. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

b. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 54.2 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C Tests and $\leq 0.75\ L_a$ for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at ≥ 8.0 psig.

   The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

   The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.21 Control Complex Habitability Envelope Integrity Program

A program shall be established to maintain the integrity of the control complex habitability envelope to ensure the dose limits of 10 CFR 50 Appendix A General Design Criteria 19 are not exceeded. The program shall establish acceptable leakage limits, ensure maintenance activities are monitored and provide a preventive maintenance program for the control complex habitability envelope.

The Control Complex Habitability Envelope Integrity Program shall ensure that:

1. Breaches in the habitability envelope are managed to ensure that in-leakage remains below design basis analysis limits.
2. The preventive maintenance program includes doors, wall/roof/floor penetrations, dampers and floor drains that are part of the control complex habitability envelope.
3. Periodic evaluations of the systems, components and key analysis assumptions are performed.
4. Configuration control of the CCHE is maintained.

BASES

LCO (continued)

b. A Control Complex Return Fan is OPERABLE;

c. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions;

d. Ductwork and dampers are OPERABLE, and air circulation can be maintained; and

e. The CCHE is intact as discussed below.

The CCHE boundary including the integrity of the doors, walls, roof, floors, floor drains, penetration seals, and ventilation isolation dampers must be maintained within the assumptions of the design calculations. Breaches in the CCHE must be controlled to provide assurance that the CCHE remains capable of performing its function.

If CCHE integrity cannot be maintained, ~~total open breach area in the CCHE exceeds the limit determined in approved design analyses (Reference 2), currently 35.5 square inches,~~ the CCHE is rendered inoperable and entry into LCO Condition B is required. ~~The upper bound of the breach area for the LCO is the sum of the breach area limit plus one square foot (144 square inches).~~ If the Required Action of LCO Condition B is not met within the respective Completion Time, then Condition C ~~or D, as applicable,~~ must be entered. ~~NOTE~~

The LCO is 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 hatches, panels and access ports, these controls consist of stationing a dedicated individual at the opening 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 isolation is indicated.

The ability to maintain temperature in the Control Complex is addressed in Technical Specification 3.7.18.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CCHE will remain habitable during and following a postulated accident. ~~During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.~~

(continued)

BASES

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore the train to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the radiation protection function for control room personnel. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

With the CCHE inoperable, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE CCHE boundary within 24 hours. During the time frame that the CCHE boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radiation, toxic chemicals and smoke. ~~due to breaches in excess of approved design calculations, but within the criteria stated, operation may continue for 7 days.~~ Restoration of the CCHE boundary ~~excess breaches~~ is not limited to returning the ~~opening~~ boundary to its ~~pre-breached~~ previous condition, but can also be accomplished using temporary sealing measures as described in plant procedures and/or work instructions.

Condition B will permit ~~opening breaches in the CCHE to support~~ maintenance and modification to the habitability envelope boundary. It also will ~~establish an allowance for the discovery of breaches during routine operation, and~~ provide the opportunity to repair the ~~breach~~ boundary in a time frame consistent with the ~~low~~ safety significance ~~of small breaches in the CCHE~~. Breaches in the envelope, that are either planned or discovered, may be evaluated in accordance with design basis documents to determine if the CCHE remains OPERABLE. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a significant release occurring during this time and the use of compensatory measures. ~~NOTI~~

~~Condition B also provides an opportunity, following an unsuccessful CCHE leak rate test, to determine the cause for excessive leakage, correct it, and perform a re-test. Excessive leakage measured during an integrated leak test can be converted to an equivalent breach size in accordance with approved design calculations. If the calculated breach size is less than or equal to 179.5 square inches then operation may continue while locating the source of the leakage and performing a re-test.~~

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

~~NOTE~~

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or CCHE boundary cannot be restored to OPERABLE status, ~~or breaches in the CCHE which exceed allowable limits cannot be closed~~ within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. ~~NOTE~~

~~D.1 and D.2~~ ~~NOTE~~

~~During movement of irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status, or breaches in the CCHE which exceed allowable limits, cannot be closed within the associated Completion Time, the OPERABLE CREVS train must immediately be placed in the emergency recirculation mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.~~ ~~NOTE~~

~~An alternative to Required Action D.1 is to immediately suspend activities that could release radioactivity and require isolation of the CCHE. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.~~ ~~NOTE~~

~~Required Action D.1 and D.2 are modified by a Note indicating to place the system in the emergency mode if automatic transfer to emergency mode is inoperable.~~

~~E.1~~D.1 ~~NOTE~~

If both CREVS trains are inoperable or ~~breaches in the CCHE exceed the limits of Condition B in MODE 1, 2, 3, or 4,~~ the CREVS may not be capable of performing the intended function and the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.4

This SR verifies that CCHE integrity is maintained. The details of the program are contained in the Control Complex Habitability Envelope Integrity Program, which is required by Technical Specification 5.6.2.21. Failure to meet individual program requirements does not necessarily make the CCHE inoperable. Each individual failure should be evaluated in accordance with design basis documents to determine if the CCHE can still perform its safety function. If the CCHE can still function as required in the design basis analysis, the system remains OPERABLE.

~~The design of the CCHE precludes performance of the commonly applied leak test characterized by pressurization to a nominal value and measurement of the make up air required to maintain pressurization. The test for CR-3 is performed by operating CREVS in the emergency recirculation mode with the Auxiliary Building Ventilation System operating to maintain a differential pressure between the CCHE and the Auxiliary Building. The Auxiliary Building will be at least 1/8 inch water gauge negative relative to the CCHE. Tracer gas will be used to determine the leakage rate. The acceptance criteria for the test is a leakage rate that would not result in control room personnel exceeding dose limits described in Reference 3 following the most limiting accident. A detailed description of the conditions for conduct of the test are provided in Reference 2.~~

~~NOTE~~

REFERENCES

1. FSAR, Section 9.7.2.1.g.

2. ~~CR-3 Control Room Habitability Report, dated July 30, 1998~~FPC Calculation N-00-0006.

3. 10 CFR 50, Appendix A, GDC 19.

4. Regulatory Guide 1.52, Rev. 2, 1978.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR - 72

ATTACHMENT C

LICENSE AMENDMENT REQUEST #262, REVISION 2
Alternative Source Term and
Control Room Emergency Ventilation System

Proposed Revised Improved Technical Specifications and Bases Change Pages

Revision Bar Format

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each CREVS train for ≥ 15 minutes.	31 days
SR 3.7.12.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.3	Verify each CREVS train actuates to the emergency recirculation mode on an actual or simulated actuation signal.	24 months
SR 3.7.12.4	Verify control complex habitability envelope integrity in accordance with ITS 5.6.2.21.	In accordance with the Control Complex Habitability Envelope Integrity Program

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

a. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

b. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 54.2 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C Tests and $\leq 0.75\ L_a$ for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 8.0 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.21 Control Complex Habitability Envelope Integrity Program

A program shall be established to maintain the integrity of the control complex habitability envelope to ensure the dose limits of 10 CFR 50 Appendix A General Design Criteria 19 are not exceeded. The program shall establish acceptable leakage limits, ensure maintenance activities are monitored and provide a preventive maintenance program for the control complex habitability envelope.

The Control Complex Habitability Envelope Integrity Program shall ensure that:

1. Breaches in the habitability envelope are managed to ensure that in-leakage remains below design basis analysis limits.
2. The preventive maintenance program includes doors, wall/roof/floor penetrations, dampers and floor drains that are part of the control complex habitability envelope.
3. Periodic evaluations of the systems, components and key analysis assumptions are performed.
4. Configuration control of the CCHE is maintained.

BASES

LCO (continued)

e. The CCHE is intact as discussed below.

The CCHE boundary including the integrity of the doors, walls, roof, floors, floor drains, penetration seals, and ventilation isolation dampers must be maintained within the assumptions of the design calculations. Breaches in the CCHE must be controlled to provide assurance that the CCHE remains capable of performing its function.

If CCHE integrity cannot be maintained, the CCHE is rendered inoperable and entry into LCO Condition B is required. If the Required Action of LCO Condition B is not met within the respective Completion Time, then Condition C must be entered.

The LCO is 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 hatches, panels and access ports, these controls consist of stationing a dedicated individual at the opening 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 isolation is indicated.

The ability to maintain temperature in the Control Complex is addressed in Technical Specification 3.7.18.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CCHE will remain habitable during and following a postulated accident.

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore the train to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the radiation protection function for control room personnel. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result

(continued)

ACTIONS (continued)

A.1 (continued)

in loss of CREVS function. The 7 day Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

With the CCHE inoperable, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE CCHE boundary within 24 hours. During the time frame that the CCHE boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radiation, toxic chemicals and smoke. Restoration of the CCHE boundary is not limited to returning the boundary to its previous condition, but can also be accomplished using temporary sealing measures as described in plant procedures and/or work instructions.

Condition B will permit maintenance and modification to the habitability envelope boundary. It also will provide the opportunity to repair the boundary in a time frame consistent with the safety significance. Breaches in the envelope, that are either planned or discovered, may be evaluated in accordance with design basis documents to determine if the CCHE remains OPERABLE. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a significant release occurring during this time and the use of compensatory measures.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or CCHE boundary cannot be restored to OPERABLE status, within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both CREVS trains are inoperable or the CREVS may not be capable of performing the intended function and the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks proper function of this system. Systems such as the CR-3 design without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with Regulatory Guide 1.52, (Ref. 4) as described in the VFTP Program description (FSAR, Section 9.7.4). The VFTP includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each CREVS train actuates to place the control complex into the emergency recirculation mode on an actual or simulated actuation signal. The Frequency of 24 months is consistent with the typical fuel cycle length.

SR 3.7.12.4

This SR verifies that CCHE integrity is maintained. The details of the program are contained in the Control Complex Habitability Envelope Integrity Program, which is required by Technical Specification 5.6.2.21. Failure to meet individual program requirements does not necessarily make the CCHE inoperable. Each individual failure should be evaluated in accordance with design basis documents to determine if the CCHE can still perform its safety function. If the CCHE can still function as required in the design basis analysis, the system remains OPERABLE.

(continued)

BASES

REFERENCES

1. FSAR, Section 9.7.2.1.g.
2. FPC Calculation N-00-0006.
3. 10 CFR 50, Appendix A, GDC 19.
4. Regulatory Guide 1.52, Rev. 2, 1978.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR - 72

ATTACHMENT D

LICENSE AMENDMENT REQUEST #262, REVISION 2
Alternative Source Term and
Control Room Emergency Ventilation System

Consolidation of LAR #262 Revisions 0, 1 and 2, Proposed Revised Improved Technical Specifications and Bases Change Pages

Revision Bar Format

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

3.7 PLANT SYSTEMS

3.7.12 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.12 Two CREVS trains shall be OPERABLE.

-----------------------------NOTE-----------------------------
The control complex habitability envelope (CCHE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
B. Two CREVS trains inoperable due to inoperable CCHE boundary.	B.1 Restore CCHE boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours
D. Two CREVS trains inoperable for reasons other than Condition B.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each CREVS train for ≥ 15 minutes.	31 days
SR 3.7.12.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.3	Verify each CREVS train actuates to the emergency recirculation mode on an actual or simulated actuation signal.	24 months
SR 3.7.12.4	Verify control complex habitability envelope integrity in accordance with ITS 5.6.2.21.	In accordance with the Control Complex Habitability Envelope Integrity Program

THIS PAGE INTENTIONALLY LEFT BLANK

3.7 PLANT SYSTEMS

3.7.18 Control Complex Cooling System

LCO 3.7.18 Two Control Complex Cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the cooling capability of a single OPERABLE Control Complex Cooling train available.	A.1 Ensure adequate cooling capability from the Control Complex Cooling system in operation. AND A.2 Restore Control Complex Cooling trains(s) to OPERABLE status.	Immediately 7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in Mode 3. AND B.2 Be in Mode 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify each chilled water pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.18.2	Verify the redundant capability of the Control Complex Cooling System to remove the assumed heat load.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.12 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The principal function of the Control Room Emergency Ventilation System (CREVS) is to provide an enclosed environment from which the plant can be operated following an uncontrolled release of radioactivity or toxic gas.

The CREVS consists of two trains with much of the non-safety related equipment common to both trains and with two independent, redundant components supplied for major items of safety related equipment (Ref. 1). The major equipment consists of the normal duty filter banks, the emergency filters, the normal duty and emergency duty supply fans, and the return fans. The normal duty filters consist of one bank of glass fiber roughing filters. The emergency filters consist of a roughing filter similar to the normal filters, high efficiency particulate air (HEPA) filters, and activated charcoal adsorbers for removal of gaseous activity (principally iodine). The rest of the system, consisting of supply and return ductwork, dampers, and instrumentation, is not designed with redundant components. However, redundant dampers are provided for isolation of the ventilation system from the surrounding environment.

The Control Complex Habitability Envelope (CCHE) is the space within the Control Complex served by CREVS. This includes Control Complex floor elevations from 108 through 180 feet and the stair enclosure from elevation 95 to 198 feet. The elements which compromise the CCHE are walls, doors, a roof, floors, floor drains, penetration seals, and ventilation isolation dampers. Together the CCHE and CREVS provide an enclosed environment from which the plant can be operated following an uncontrolled release of radioactivity or toxic gas.

CREVS has a normal operation mode and recirculation modes. During normal operation, the system provides filtered, conditioned air to the control complex, including the controlled access area (CA) on the 95 foot elevation. When switched to the recirculation mode, isolation dampers close isolating the discharge to the controlled access area and isolating the outside air intake. In this mode the system recirculates filtered air through the CCHE.

(continued)

BASES

BACKGROUND (continued)

The control complex normal duty ventilation system is operated from the control room and runs continuously. During normal operation, the outside air intake damper is partially open, the atmospheric relief discharge damper is closed, the discharge to the CA is open, and the system return damper is throttled. This configuration allows a controlled amount of outside air to be admitted to the control complex. The design temperature maintained by the system is 75°F at a relative humidity of 50%.

Two signals will cause the system to automatically switch to the recirculation modes of operation.

1. Engineered Safeguards Actuation System (ESAS) signal (high reactor building pressure).

2. High radiation signal from the return duct radiation monitor RM-A5.

The recirculation modes isolate the CCHE from outside air to ensure a habitable environment for the safe shutdown of the plant. In these modes of operation, the controlled access area is isolated from the CCHE.

Upon detection of ESAS, the system switches to the normal recirculation mode. In this mode, dampers for the outside air intake and the exhaust to the CA will automatically close, isolating the CCHE from outside air exchange, and the system return damper will open thus allowing air in the CCHE to be recirculated. Additionally, the CA fume hood exhaust fan, CA fume hood auxiliary supply fan, and CA exhaust fan are de-energized and their corresponding isolation dampers close. The return fan, normal filters, normal fan, and the cooling (or heating) coils remain in operation in a recirculating mode.

Upon detection of high radiation by RM-A5 the system switches to the emergency recirculation mode. In this mode, the dampers that isolate the CCHE from the surroundings will automatically close. The CA fume hood exhaust fan, CA fume hood auxiliary supply fan, CA exhaust fan, normal supply fan, and return fan are tripped and their corresponding isolation dampers close. Manual action is required to restart the return fan and place the emergency fans and filters in operation. The cooling (or heating) coils remain in operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

During emergency operations the design basis of the CREVS and the CCHE is to provide radiation protection to the control room operators. The limiting accident which may threaten the habitability of the control room (i.e., accidents resulting in release of airborne radioactivity) is the postulated Control Rod Ejection accident. The consequences of this event result in the limiting radiological source term for the control room habitability evaluation (Ref. 2). The CREVS and the CCHE ensures that the control room will remain habitable following all postulated design basis events, maintaining exposures to control room operators within the limits of GDC 19 of 10 CFR 50 Appendix A (Ref. 3).

The CREVS is not in the primary success path for any accident analysis. However, the Control Room Emergency Ventilation System meets Criterion 3 of the NRC Policy Statement since long term control room habitability is essential to mitigation of accidents resulting in atmospheric fission product release.

LCO

Two trains of the control room emergency ventilation system are required to be OPERABLE to ensure that at least one is available assuming a single failure disabling the other train. Failure to meet the LCO could result in the control room becoming uninhabitable in the unlikely event of an accident.

The required CREVS trains must be independent to the extent allowed by the design which provides redundant components for the major equipment as discussed in the BACKGROUND section of this bases. OPERABILITY of the CREVS requires the following as a minimum:

a. A Control Complex Emergency Duty Supply Fan is OPERABLE;

b. A Control Complex Return Fan is OPERABLE;

c. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions;

d. Ductwork and dampers are OPERABLE, and air circulation can be maintained; and

(continued)

BASES

LCO (continued)

e. The CCHE is intact as discussed below.

The CCHE boundary including the integrity of the doors, walls, roof, floors, floor drains, penetration seals, and ventilation isolation dampers must be maintained within the assumptions of the design calculations. Breaches in the CCHE must be controlled to provide assurance that the CCHE remains capable of performing its function.

If CCHE integrity cannot be maintained, the CCHE is rendered inoperable and entry into LCO Condition B is required. If the Required Action of LCO Condition B is not met within the respective Completion Time, then Condition C must be entered.

The LCO is 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 hatches, panels and access ports, these controls consist of stationing a dedicated individual at the opening 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 isolation is indicated.

The ability to maintain temperature in the Control Complex is addressed in Technical Specification 3.7.18.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CCHE will remain habitable during and following a postulated accident.

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore the train to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the radiation protection function for control room personnel. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result

(continued)

BASES

ACTIONS (continued)

A.1 (continued)

in loss of CREVS function. The 7 day Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

With the CCHE inoperable, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE CCHE boundary within 24 hours. During the time frame that the CCHE boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radiation, toxic chemicals and smoke. Restoration of the CCHE boundary is not limited to returning the boundary to its previous condition, but can also be accomplished using temporary sealing measures as described in plant procedures and/or work instructions.

Condition B will permit maintenance and modification to the habitability envelope boundary. It also will provide the opportunity to repair the boundary in a time frame consistent with the safety significance. Breaches in the envelope, that are either planned or discovered, may be evaluated in accordance with design basis documents to determine if the CCHE remains OPERABLE. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a significant release occurring during this time and the use of compensatory measures.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or CCHE boundary cannot be restored to OPERABLE status, within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both CREVS trains are inoperable the CREVS may not be capable of performing the intended function and the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks proper function of this system. Systems such as the CR-3 design without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with Regulatory Guide 1.52, (Ref. 4) as described in the VFTP Program description (FSAR, Section 9.7.4). The VFTP includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each CREVS train actuates to place the control complex into the emergency recirculation mode on an actual or simulated actuation signal. The Frequency of 24 months is consistent with the typical fuel cycle length.

SR 3.7.12.4

This SR verifies that CCHE integrity is maintained. The details of the program are contained in the Control Complex Habitability Envelope Program, which is required by Technical Specification 5.6.2.21. Failure to meet individual program requirements does not necessarily make the CCHE inoperable. Each individual failure should be evaluated in accordance with design basis documents to determine if the CCHE can still perform its safety function. If the CCHE can still function as required in the design basis analysis, the system remains OPERABLE.

(continued)

BASES

REFERENCES

1. FSAR, Section 9.7.2.1.g.
2. FPC Calculation N-00-0006.
3. 10 CFR 50, Appendix A, GDC 19.
4. Regulatory Guide 1.52, Rev. 2, 1978.

THIS PAGE INTENTIONALLY LEFT BLANK

B 3.7 PLANT SYSTEMS

B 3.7.18 Control Complex Cooling System

BASES

BACKGROUND

The Control Complex Cooling System provides temperature control for the control room and other portions of the Control Complex containing safety related equipment.

The Control Complex Cooling System consists of two redundant chillers, associated chilled water pumps, and parallel duct mounted air heat exchangers that can receive chilled water from either chilled water pump. A train consists of a chiller and associated chilled water pump as well as a duct mounted heat exchanger that provide cooling of recirculated control complex air. The design of the Control Complex Cooling System contains features that allow either train chiller and associated chilled water pump to provide cooling capability to either duct mounted heat exchanger. Redundant chillers and chilled water pumps are provided for suitable temperature conditions in the control complex for operating personnel and safety related control equipment. The Control Complex Cooling System maintains the nominal temperature between 70°F and 80°F.

A single chiller and associated chilled water pump will provide the required heat removal for either duct mounted heat exchanger. The Control Complex Cooling System operation to maintain control complex temperature is discussed in the FSAR, Section 9.7 (Ref. 1).

APPLICABLE SAFETY ANALYSIS

The Control Complex Cooling System consists of redundant, safety related components, with some common piping. The Control Complex Cooling System maintains the temperature between 70°F and 80°F. A single active failure of a Control Complex Cooling System component does not impair the ability of the system to perform as designed. The Control Complex Cooling System is designed in accordance with Seismic Category I requirements. The Control Complex Cooling System is capable of removing heat loads from the control room and other portions of the Control Complex containing safety related equipment, including consideration of equipment heat loads and

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)	personnel occupancy requirements, to ensure equipment OPERABILITY. The Control Complex Cooling System satisfies Criterion 3 of the NRC Policy Statement.
LCO	Two redundant trains of the Control Complex Cooling System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure disables one redundant component. A Control Complex Cooling train consists of a chiller and associated chilled water pump as well as a duct mounted heat exchanger that provides cooling of recirculated control complex air. All components of an OPERABLE train must be energized by the same train electrical bus. Total system failure could cause control complex equipment to exceed its operating temperature limits. In addition, the Control Complex Cooling System must be OPERABLE to the extent that air circulation can be maintained (See Specification 3.7.12).
APPLICABILITY	In MODES 1, 2, 3, and 4, the Control Complex Cooling System must be OPERABLE to ensure that the control complex temperature will not exceed equipment OPERABILITY requirements.
ACTIONS	A.1 The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy and diversity of subsystems, the inoperability of one component in a train does not render the Control Complex Cooling System incapable of performing its safety function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the Control Complex Cooling System. The intent of this Condition is to maintain a combination of equipment such that the cooling capability equivalent to 100% of a single train remains available and in operation. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

(continued)

BASES

ACTIONS

A.1 (continued)

With one or more components inoperable such that the cooling capability equivalent to a single OPERABLE train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

With one or more Control Complex Cooling trains inoperable and at least 100% cooling capability of a single OPERABLE train available, the inoperable components must be restored to OPERABLE status within 7 days. In this Condition, the remaining Control Complex Cooling System equipment is adequate to maintain the control complex temperature. Adequate cooling capability exists when the control complex air temperature is maintained within the limits for the contained equipment and components. However, the overall reliability is reduced because additional failures could result in a loss of Control Complex Cooling System function. The 7 day Completion Time is based on the low probability of an event occurring requiring the Control Complex Cooling System and the consideration that the remaining components can provide the required capabilities.

B.1 and B.2

If the inoperable Control Complex Cooling System component cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

Verifying that each Control Complex Cooling chiller's developed head at the flow test point is greater than or equal to the required developed head ensures that chiller's performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 3). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.7.18.2

This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. A 24 month Frequency is appropriate, as significant degradation of the system is slow and is not expected over this time period.

REFERENCES

1. FSAR, Section 9.7.
2. Deleted.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

5.6 Procedures, Programs and Manuals

5.6.2.11 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.6.2.12 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of the Control Room Emergency Ventilation System (CREVS) per the requirements specified in Regulatory Guide 1.52, Revision 2, 1978, and/or as specified herein, and in accordance with ANSI N510-1975 and ASTM D 3803-89 (Re-approved 1995).

a. Demonstrate for each train of the CREVS that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, 1978, and in accordance with ANSI N510-1975 at the system flowrate of between 37,800 and 47,850 cfm.

b. Demonstrate for each train of the CREVS that an inplace test of the carbon adsorber shows a system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate of between 37,800 and 47,850 cfm.

c. Demonstrate for each train of the CREVS that a laboratory test of a sample of the carbon adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, 1978, meets the laboratory testing criteria of ASTM D 3803-89 (Re-approved 1995) at a temperature of 30°C and relative humidity of 95% with methyl iodide penetration of less than 5.0%.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.12 VFTP (continued)

d. Demonstrate for each train of CREVS that the pressure drop across the combined roughing filters, HEPA filters and the carbon adsorbers is $\leq \Delta P = 4"$ water gauge when tested in accordance with Regulatory Guide 1.52, Revision 2, 1978, and ANSI N510-1975 at the system flowrate of between 37,800 and 47,850 cfm.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.6.2.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Radioactive Waste Disposal (WD) System, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Radioactive Waste Disposal (WD) System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria, (i.e., whether or not the system is designed to withstand a hydrogen explosion).

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.14 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has the following properties within limits of ASTM D 975 for Grade No. 2-D fuel oil:

    1. Kinematic Viscosity,

    2. Water and Sediment,

    3. Flash Point,

    4. Specific Gravity API;

b. Other properties of ASTM D 975 for Grade No. 2-D fuel oil are within limits within 92 days following sampling and addition of new fuel to storage tanks.

c. Total particulate contamination of stored fuel oil is < 10 mg/L when tested once per 92 days in accordance with ASTM D 2276-91 (gravimetric method).

5.6.2.15 Not Used

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

a. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

b. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 54.2 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C Tests and $\leq 0.75\ L_a$ for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 8.0 psig.

   The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

   The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.21 Control Complex Habitability Envelope Integrity Program

A program shall be established to maintain the integrity of the control complex habitability envelope to ensure the dose limits of 10 CFR 50 Appendix A General Design Criteria 19 are not exceeded. The program shall establish acceptable leakage limits, ensure maintenance activities are monitored and provide a preventive maintenance program for the control complex habitability envelope.

The Control Complex Habitability Envelope Integrity Program shall ensure that:

1. Breaches in the habitability envelope are managed to ensure that in-leakage remains below design basis analysis limits.
2. The preventive maintenance program includes doors, wall/roof/floor penetrations, dampers and floor drains that are part of the control complex habitability envelope.
3. Periodic evaluations of the systems, components and key analysis assumptions are performed.
4. Configuration control of the CCHE is maintained.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR - 72

APPENDIX A

LICENSE AMENDMENT REQUEST #262, REVISION 2
Alternative Source Term and
Control Room Emergency Ventilation System

Toxic Gas Report

Toxic Gas Report

Florida Power Corporation

Crystal River – Unit 3

August 2001

Table of Contents

I.	Introduction and Summary	3
II.	Control Complex Habitability Envelope and Control Complex Ventilation System Description	3
III.	Toxic Gas History and Evaluation	3
IV.	References	5

I. Purpose

The Crystal River, Unit 3 (CR-3) Operating License contains a requirement to maintain control room habitability as specified in the post-Three Mile Island (TMI) requirements of NUREG-0737. One aspect of Control Room Habitability is to ensure the operators can remain in the control room and are functional following a toxic gas release. The purpose of this report is to provide information related to the ability of CR-3 to meet this requirement.

II. Control Complex Habitability Envelope and Control Complex Ventilation System Description

The CCHE is the physical barrier that separates the control room environment from the external environment. The integrity of the CCHE barriers (walls, doors, ceilings, floors, sealed penetrations, ventilation penetrations, etc.) helps minimize the intake of toxic gas following ventilation system isolation.

The Control Complex is a six floor building located between the auxiliary building and the turbine building. The CCHE is the top five floors of the Control Complex. The lower floor is isolated from the CCHE under toxic gas conditions. The top floor of the CCHE contains the control complex ventilation equipment, thus it is all internal to the CCHE. The control room is one floor below the ventilation equipment room.

Control Complex normal ventilation maintains a continuous supply and exhaust flow. Upon operator detection of toxic gas in the control complex (nasal detection or other physical effects), plant procedures instruct the operator to isolate the normal supply and exhaust and place the control complex ventilation in the recirculation mode. The recirculation mode provides environmental control for personnel comfort and equipment operation and protection of control room personnel during toxic gas events. The control complex is not pressurized to limit inleakage.

III. Toxic Gas History and Evaluation

Regulatory Guide 1.78 provides information and assumptions for assessing toxic gas accidents with regard to control room habitability. From this document comes the basic criteria that, in the event of a toxic gas accident, appropriate toxicity limits not be exceeded in the control room two minutes after initial detection in order to allow the operator adequate time to take action (i.e., don an air pack) prior to being overcome. The Regulatory Guide allows for detection to be accomplished by personal means (nasal detection) or with automatic detection equipment. Ventilation system isolation, if required, can be attained either by operator action or by an automatic signal from toxic gas detectors. At CR-3, procedures provide the appropriate instructions for the operator in the event of a toxic gas accident and for the use of air packs.

Based on previous evaluations, the locations and quantities of toxic gas storage sites at the Crystal River (CR) site that posed a potential liability to CR-3 control room habitability are listed below:

Toxic Gas Container Size and Location

Toxic Gas	Helper Cooling Towers	CR-1/CR-2	CR-4/ CR-5
Chlorine -- Cl_2	17 tons	none	1 ton cylinders
Sulfur Dioxide --SO_2	50 tons	45 tons	1 ton cylinders

The most limiting source of toxic gas was an SO_2 tank at CR-1, which had been administratively limited to storage of 30 tons. Automatic detection and isolation was required as a result of that tank. That tank has been replaced with a system that uses solid pellets, which are converted to SO_2 as needed. The tank has been emptied of its contents and will no longer be used. As such, it is no longer the limiting source.

The next most limiting toxic gas source is the Helper Cooling Towers. Currently, there is no SO_2 or Cl_2 stored at this location. If chemicals were stored at this location, administrative controls are in place to limit the Helper Cooling Towers to 3 tons of Cl_2 and 17 tons of SO_2. FPC has used revised calculations to evaluate the lower quantities: 3 tons of Cl_2 and 17 tons of SO_2. This analysis concludes that CREVS could remain in its normal alignment (i.e., no CCHE isolation required) without exceeding Control Room toxicity limits within 2 minutes of nasal detection.

In May 2001, the one-ton cylinders of gaseous Cl_2 and SO_2 were removed from CR Units 4/5. CR Units 4/5 now have a system that uses sodium hypochlorite and sodium bisulfite. This also eliminated any frequent transport of toxic chemicals to the site.

Based on the removal of all significant sources of toxic gas, and the administrative controls on the quantity allowed at the Helper Cooling Towers, the CR-3 control room SO_2 and chlorine detectors and their auto-isolation functions have been removed.

Therefore, due to the limited quantity of toxic gas on site, CR-3 is able to meet the criteria of Regulatory Guide 1.78 by donning of respiratory protection equipment within 2 minutes followed by subsequent manual isolation of normal ventilation.

IV. References

1. FPC Calculation No. M97-0109 R/1 (SL-9929-M-0008 R/1), Toxic Gas Analysis
2. NRC Regulatory Guide 1.78 – Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, June 1974

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR - 72

APPENDIX B

LICENSE AMENDMENT REQUEST #262, REVISION 2
Alternative Source Term and
Control Room Emergency Ventilation System

Revised Summary of Radiological Analyses for FSAR Chapter 14 Accidents

Summary of Radiological Analyses for FSAR Chapter 14 Accidents

Each of the Design Basis Accidents in Chapter 14 that had a quantified calculation of the dose consequences was recalculated using the Alternative Source Term where applicable. All dose results are expressed in terms of TEDE for comparison with the limits of 10CFR50.67. Unless noted, the accident consequences were calculated to both the Control Room Operator and the public at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ). The primary method used for these analyses was the NRC developed code, RADTRAD, Version 3.02.

The following is a summary of the scenario, analysis and results for each of the accidents. The detailed inputs to the calculation and their bases are provided in a format consistent with the order in which they are entered into the RADTRAD input files. This input information is provided as individual Appendices in this submittal.

1. Main Steam Line Break – FSAR Section 14.2.2.1

   Scenario – For the purpose of radiological analysis, this accident is assumed to involve the rupture of a main steam line outside containment, but upstream of the MSIV's. The release consists of complete loss of the secondary coolant plus additional feedwater from the affected generator, secondary coolant from the unaffected generator, and a continuing primary to secondary leak in the affected generator of 1 gpm for 85 hours. The reactor coolant activity is based on steady state operation with 1% degraded fuel cladding. No spiking is assumed. This concentration is higher than allowed by Technical Specifications. There is no calculated fuel damage as a result of the accident.

   Analysis - There are no changes to the plant design or procedures associated with this license amendment request that would affect the current dose assessment. There are no aspects of the source term specified in Regulatory Guide 1.183 that would affect the existing dose calculation, as there is no fuel damage associated with this accident. The only aspect of the current Chapter 14 analysis affected by 10CFR50.67 and CR-3's intent to apply a Full Application of the Alternative Source term is the requirement to express the dose results of the existing calculation in terms of TEDE. This is required to demonstrate compliance with the limits of 10CFR50.67. Therefore, a new RADTRAD analysis for this accident was not required. A calculation was performed using the dose conversion factors in RADTRAD to convert the current dose results to TEDE results. Appendix C provides the inputs used in the current analysis and the conversion to TEDE results.

   Results:

   EAB TEDE = 0.0047 REM (0-2 hr is worst 2 hr period)
   LPZ TEDE = 0.0031 REM

The releases for this accident are significantly less than for a Steam Generator Tube Rupture (SGTR). Therefore, a control room dose assessment was not performed for this accident, as it would be bounded by the SGTR.

2. Steam Generator Tube Rupture – FSAR Section 14.2.2.2

Scenarios - There are two scenarios analyzed for the SGTR. They are the licensing basis analysis for public dose and the informational Standard Review Plan analysis performed for control room habitability.

The licensing basis scenario for public dose assumes no single failure and no Loss of Offsite Power (LOOP). Without a LOOP, the condenser remains available and iodine releases are significantly reduced. No iodine spike cases are analyzed. Significant conservatism is applied to the assumed release from the primary side, as discussed below.

In response to a request for additional information from the NRC related to control room habitability, in 1998, CR-3 analyzed the control room dose for a SGTR scenario consistent with the guidelines of the Standard Review Plan (Iodine spiking, LOOP, single failure).

The assumed licensing basis scenario is:

T=0 – Tube rupture. The primary to secondary break flow is conservatively calculated to be 60.5 lbm/sec (includes an existing 1 gpm primary to secondary leak) and is assumed to continue for the entire 8 hours of the event.

T= 8 min – the reactor trips. Until this time, all releases are via the condenser. With the reactor/turbine trip, a fraction of the steam is released via the atmospheric dumps and main steam safeties. The fraction of steam that is released directly to atmosphere from 8 minutes to 9 minutes is 0.55.

T= 9 minutes – steam flow is such that the dumps and safeties are closed. All releases for the remainder of the accident are via the condenser. The assumed reduction of iodine in the condenser is a factor of 10,000. In regard to noble gas releases, the release path doesn't matter. All noble gases released from the primary side are assumed to be instantaneously released to the environment.

T= 8 hours – the plant has been cooled to the point where both steam generators are isolated and heat removal is via the Decay Heat system. This terminates the release.

The reactor coolant activity for the licensing basis scenario is based on steady state operation with 1% degraded fuel cladding. No spiking is assumed. This concentration is higher than allowed by Technical Specifications. Releases of secondary side activity are not included in the calculation due to their insignificance compared to the primary side releases.

The control room dose SRP-like analysis considers a LOOP, the most-limiting single failure, and both a pre-existing iodine spike at the maximum value allowed in the Technical Specification (60 μCi/gm dose equivalent I-131) and an iodine spike 500 times the equilibrium release rate coincident with the operation value of 1 μCi/gm dose equivalent I-131. Credit is taken for some non-safety-related components, specifically, the analysis credits the use of the ADVs for cooldown, and the use of the PORV for depressurization.

The sequence of events for the SRP-like SGTR analysis, including control room assumptions, is as follows:
T = 0 - Tube Rupture
T = 8 min. - Rx trip, LOOP, loss of condenser, isolation of CCHE with LOOP
T = 38 min. - CR ventilation on recirculation 30 min. after LOOP
T = 53 min. - Operators initiate cooldown with ADV's – ADV fails on unaffected generator
T = 55 min. - Initiate RCS depressurization with PORV's
T = 78 min. - Manually open failed ADV
T = 108 min. - Isolate affected OTSG
T = 8 hrs - Terminate release from unaffected generator

The analysis conservatively considered a LOOP coincident with the reactor trip. The limiting single failure is the failure of the Atmospheric Dump Valve (ADV) on the unaffected steam generator to automatically open in response to the post-trip pressure control signal. This forces the operator to cool down using the affected generator's ADV until the unaffected steam generator's ADV can be opened. This results in an extended cooldown time and hence time until the affected generator can be isolated.

Analysis - For the Chapter 14 licensing basis assessment, there are no changes to the plant design or procedures associated with the planned Alternative Source Term license amendment request that would affect the current dose assessment. There are no aspects of the source term specified in Reg. Guide 1.183 that would affect the existing dose calculation, as there is no fuel damage associated with this accident. The only aspect of the current Chapter 14 analysis affected by 10CFR50.67 and CR-3's intent to apply a Full Application of the Alternative Source term is the requirement to express the dose results of the existing calculation in terms of TEDE dose. This is required to demonstrate compliance with the limits of 10CFR50.67. A calculation was performed using the dose conversion factors in RADTRAD to convert the current dose results to TEDE results. Appendix D provides the inputs used in the current analysis and the conversion to TEDE results.

For the SRP-like SGTR analysis of control room dose, there are changes associated with the proposed license amendment that will affect the calculation of control room dose. Therefore, a reanalysis of the control room dose using RADTRAD has been performed. For informational purposes, the public dose results will also be presented for the SRP-like scenario. Appendix E presents the RADTRAD inputs and their bases.

Results:

Chapter 14 Design Basis Assumptions

EAB TEDE – 0.139 REM (0-2 hr is worst 2 hour period)
LPZ TEDE – 0.0455 REM

SRP Assumptions

EAB TEDE -Pre-accident spike – 5.98 REM
EAB TEDE -Post-accident spike – 2.40 REM
LPZ TEDE - Pre-accident spike – 0.523 REM
LPZ TEDE - Post-accident spike – 0.210 REM
Control Room TEDE -Pre-accident spike – 1.19 REM
Control Room TEDE -Post-accident spike – 0.365 REM

3. Fuel Handling Accident – FSAR Section 14.2.2.3

Scenario - The gap activity is conservatively assumed to be released from all fuel pins in one fuel assembly. The gap fractions from Regulatory Guide 1.183 are used. A reduction factor for iodine is applied for removal by the water in the reactor cavity or spent fuel pool. No credit is taken for either building isolation or release pathway filters. As such, the calculation of a FHA in the spent fuel pool area will be exactly the same as for an accident in the Reactor Building. Hence, only one calculation is performed. All the activity is conservatively assumed to be released within the first few minutes. The control complex is assumed to be in the normal ventilation mode. No credit is taken for automatic isolation by the control room ventilation radiation monitor (RM-A5) or for manual isolation by the operator. No credit is taken for control complex recirculation filters. If the radiation monitor or operator did isolate the control complex and place the ventilation in filtered recirculation, the doses would be less than the results of this calculation.

Analysis – The doses were calculated using RADTRAD. Appendix F presents the RADTRAD inputs and their bases.

Results:

EAB TEDE (0-2 hr is worst 2 hour period) - 0.83 REM
LPZ TEDE - 0.073 REM
Control Room TEDE - 4.43 REM

4. Control Rod Ejection Accident – FSAR Section 14.2.2.4.

Scenario - There are two cases analyzed for the Control Rod Ejection (CRE). The first case assumes that all the activity released from the failed fuel is released directly to the containment and then leaks from the containment to the environment. The second case assumes that all the activity released from the fuel is dispersed in the reactor coolant and then is released to the environment via primary to secondary leakage.

Thermal hydraulic calculations determined that 14% of the fuel is assumed to experience clad damage (FTI 51-1172602-00). (This was very conservatively determined by assuming that any core area experiencing DNB resulted in clad damage). No fuel was calculated to experience fuel melt. Gap activity is assumed to be released from this 14% of the fuel. The gap activity is taken from Reg. Guide 1.183, Appendix H as 10% of the iodines and noble gases. Since less than 100% of the core is affected, the activity is multiplied by the maximum allowed radial peaking factor of 1.8. Normal coolant activity is insignificant compared to the activity from the 14% of the fuel that fails and hence the normal coolant activity is not included in the calculation.

For the first case, all the activity released from the fuel is instantaneously dispersed to the containment. There is no credit taken for removal by sprays. No immediate plateout is assumed. The only removal mechanisms assumed are natural deposition for the aerosol iodine and decay for all nuclides. The containment is assumed to leak at the Technical Specification leak rate limit for 24 hours and at one half that limit for the remaining 29 days.

For the second case, all the activity released from the fuel is instantaneously dispersed in the reactor coolant. The reactor coolant is assumed to leak at 150 gpd in each steam generator (300 gpd total). All activity released to the secondary side is assumed to be immediately released to the environment. It is assumed that there is a loss of offsite power and hence the condenser is not available. There is no credit assumed for any partitioning factor or removal of iodine or other nuclides in the secondary side. It is also conservatively assumed that the plant continues to cool down by steaming through the steam generators for 72 hours prior to going on the Decay Heat System.

Analysis – The doses were calculated using RADTRAD. Appendix G, Rev. 1 presents the RADTRAD inputs and their bases for the containment release scenario. Appendix H, Rev. 1 presents the RADTRAD inputs and their bases for the secondary side release scenario.

Results:

Containment Release
- EAB TEDE – 1.03 REM (0-2hr is worst 2 hours)
- LPZ TEDE – 0.288 REM
- Control Room TEDE – 0.754 REM

Secondary Side Release
- EAB TEDE – 2.10 REM (0-2hr is worst 2 hours)
- LPZ TEDE – 0.819 REM
- Control Room TEDE – 3.49 REM

Additionally, the CRE, secondary side release, is the bounding dose for CCHE/CREVS operability. Therefore, the margin between the CRE dose and the 10CFR50.67 limit of 5 REM TEDE will be used to establish the breach margin for the CCHE.

Controlled breaches are allowed to be opened in the Control Complex Habitability Envelope to a size that would ensure that the dose remains within the 5 REM TEDE acceptance criterion.

RADTRAD runs were made varying the control complex inleakage until a total dose of 5 REM was achieved. These calculations determined that an additional 400 cfm from controlled breaches would be acceptable. Breach limits and controls will be specified in the Control Complex Habitability Envelope Integrity Program.

5. LOCA – FSAR Section 14.2.2.5

Scenario/Analysis - There are a number of different contributors to the LOCA dose assessment. They are:

- The dose from leakage of recirculated ECCS water. This source is considered for both the public and control room operator. Appendix I presents the input assumptions for this analysis. The calculations are performed using the NRC Code RADTRAD.
- The dose from leakage of containment atmosphere. This source is considered for both the public and control room operator. Appendix J presents the input assumptions for this analysis. The calculations are performed using the NRC Code RADTRAD.
- The dose from an assumed containment purge at 14 days into the accident for hydrogen control. This source is considered for both the public and control room operator. Appendix K presents the input assumptions for this analysis. The calculations are performed using the NRC Code RADTRAD
- Direct doses from containment shine, charcoal filters in the Control Complex and the plume outside the control room. These direct doses are considered in the control room dose analysis and are discussed below.

The control room dose from sources outside the control room will all be direct gamma dose and hence will be a Deep Dose Equivalent (DDE). Therefore, the calculated dose can be added directly to the TEDE from activity inside the control room to get a total TEDE.

The dose due to the plume outside the control room was calculated in M-97-0110, Rev 4 to be 0.01 REM. Based on the reduced magnitude and delayed timing of the AST compared to the TID-14844 source term, the plume dose rate will be less for the AST. Assumptions such as the containment leak rate have not been changed that would affect this conclusion. It is therefore conservative to assume that the direct dose in the control room from the plume shine from outside the control room is the same as calculated in M-97-0110, or 0.01 REM.

The dose from containment shine was calculated in M-97-0110 to be 0.03 REM. Since the plume whole body dose from containment leakage is directly related to the containment airborne concentration, the above logic is also applicable to this source. The TID contribution from containment shine will be greater than the AST contribution. Therefore, it is conservative to use the dose of 0.03 REM as calculated in M-97-0110 for this AST analysis.

The dose from the control complex charcoal filters was calculated in M-97-0110 to be 0.026 REM. In this case, a number of changes have been made to assumptions that affect the amount of iodines and particulates taken into the control complex (e.g.- control complex inleakage rate). The changes are such that the integrated activity taken into the control complex is greater for the AST analysis. Therefore, it is not conservative to use the dose from M-97-0110. An approximate measure of the integrated activity in the control complex over 30 days, and hence eventually on the control complex filters, can be determined by evaluating the 30 day thyroid dose. For the TID analysis in M-97-0110, the 30 day operator thyroid dose is approximately 19 REM. For the AST analysis, the thyroid dose is approximately 47.4 REM, or approximately 2.5 times greater. Other factors, such as the effects of other particulates besides iodine can also have an effect. However, the calculation of direct dose from the filters in M-97-0110 is very conservative as it only used a concrete thickness of 1 foot between the filters on the 164' level of the control complex and the control room on the 145' level. The actual shielding thickness is a minimum of 2 feet of concrete. This extra foot of concrete would reduce the dose by at least a factor of 10. This factor of 10 would more than compensate for the effects of other nuclides or other minor differences. Therefore, the direct dose from the filters for the AST case will conservatively be estimated as 2.5 x 0.026 REM = 0.07 REM.

Results:

EAB TEDE – 7.59 REM (0.8 to 2.8 hrs is worst 2 hour period)
LPZ TEDE = 1.07 REM
Control Room TEDE = 2.30 REM

6. Letdown Line Rupture – FSAR Section 14.2.2.6

Scenario - At time zero, the letdown line outside containment is postulated to break releasing primary coolant in the Auxiliary Building. For the first six minutes, control room personnel try to keep the reactor at full power. During this time period, the control room is still assumed to be in normal makeup, with 5700 cfm of outside air. No credit is taken for the automatic radiation monitor isolation. No credit is taken for the Auxiliary Building filters in reducing the release to the environment.

At approximately six minutes, the reactor trips. At this time, there are two scenarios analyzed. The first assumes that with the reactor trip, simultaneously, there is a Loss Of Offsite Power (LOOP). As a result of the LOOP, control room ventilation trips resulting in no makeup or recirculation of the air within the control room. Because there is a LOOP, the Auxiliary Building Ventilation System loses power, however, this does not factor into the dose scenarios as no credit had been assumed for the

Auxiliary Building filters. At this time, unfiltered inleakage into the control room begins.

The second scenario assumes that with the reactor trip at approximately 6 minutes, there is no simultaneous LOOP. Therefore, the control room ventilation continues to operate with a normal makeup of 5700 cfm. The Auxiliary Building ventilation would continue to operate, however, this does not factor into the dose scenarios as no credit had been assumed for the Auxiliary Building filters.

The accident continues with these conditions until 19.5 minutes. At this time, control room personnel recognize and isolate the letdown line break ending the release.

For the scenario with the LOOP, it was assumed that the control room recirculation filters are started 30 minutes after the LOOP; therefore, at 36 minutes into the accident, the control room filtered recirculation is started. From this time, all conditions remain the same until the end of the dose calculation scenario (30 days). For the scenario without the LOOP, the control room is assumed to operate in the normal ventilation mode (5700 cfm intake, no recirculation filters) until the end of the dose calculation period (30 days).

The release rate calculations evaluated two scenarios in accordance with Standard Review Plan 15.6.2. The two scenarios are the accident initiated spike and the pre-accident spike cases. Therefore, there are a total of four scenarios:

Case 1 – LOOP / Pre-accident spike
Case 2 – LOOP / Post-accident spike
Case 3 – No LOOP / Pre-accident spike
Case 4 – No LOOP / Post-accident spike

The activity release rate into the auxiliary building was calculated by FTI and presented in FTI Document 86-1266335-00, "CR-3 Letdown Line Break SRP Activities." The activity release rates are based on the following:

- The thermal hydraulic calculations of M-96-0043 for the break flow rates.
- For the post-accident spike case - an equilibrium reactor coolant concentration of noble gases and iodines corresponding to the Technical Specification limit of 1 μCi/gm Dose Equivalent I-131, followed by a spike factor of 500 in the steady state iodine release rate from the fuel.
- For the pre-accident spike case – a reactor coolant iodine concentration corresponding to the Technical Specification short-term concentration limit of 60 μCi/gm Dose Equivalent I-131 and a noble gas concentration corresponding to the 1 μCi/gm Dose Equivalent I-131 limit.
- An assumed carryover fraction from the RCS liquid to the Auxiliary Building atmosphere of 100% of the noble gases and 10% of the iodines.

Analysis – The analysis was performed with RADTRAD. Appendix L presents the input assumptions and their bases.

Results:

EAB TEDE (0-2 hr is worst 2 hour period)
Pre-accident spike – 0.614 REM
Post-accident spike – 0.078 REM

LPZ TEDE
Pre-accident spike – 0.054 REM
Post-accident spike – 0.0068 REM

Control Room TEDE – LOOP – CR Isolation/Filtration
Pre-accident spike – 0.895 REM
Post-accident spike – 0.060 REM

Control Room TEDE – no LOOP – no CR Isolation/Filtration
Pre-accident spike – 3.24 REM
Post-accident spike – 0.339 REM

7. Waste Gas Decay Tank Rupture – FSAR Section 14.2.2.8

Scenario - The entire contents of all three waste gas decay tanks is assumed to be released. Each tank is assumed to be at its maximum allowed activity as specified in the Offsite Dose Calculation Manual. This is a very conservative assumption, as the activity in two tanks would have significantly decayed while the third tank was being filled. Additionally, the tanks are isolated from each other and it is only credible to fail one tank. The activity released from the tanks is assumed to be released to the environment instantaneously. There is no credit for mitigation features such as filters.

Analysis – The dose to the public was analyzed using RADTRAD. The input assumptions and their bases are provided in Appendix M. The releases from this accident are small compared to other accidents, so the control room dose is not analyzed.

Results:

EAB TEDE – 0.125 REM (0-2 hr is worst 2 hr period)
LPZ TEDE – 0.011 REM