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

Ref: 10 CFR 50.90

April 25, 2013
3F0413-01

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

Subject: Crystal River Unit 3 –License Amendment Request #313, Revision 0, Revision to Improved Technical Specifications Administrative Controls for Permanently Defueled Conditions

References:
1. NRC to FPC letter dated March 13, 2013, "Crystal River Unit 3 Nuclear Generating Plant Certification of Permanent Cessation of Operation and Permanent Removal of Fuel from the Reactor" (ADAMS Accession No. ML13058A380)
2. FPC to NRC letter dated April 15, 2013, "Crystal River Unit 3 – Request for Approval of the Certified Fuel Handler Training and Retraining Program"

Dear Sir:

Pursuant to 10 CFR 50.90, Florida Power Corporation (FPC) hereby provides this License Amendment Request (LAR) to revise portions of Section 5.0, Administrative Controls, of the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS).

In Reference 1, the NRC acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessel. Accordingly, pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel. The basis for this LAR is that certain Administrative Controls in the current CR-3 ITS may be revised or removed for permanently defueled conditions.

Part of this request proposes changes to the staffing and training requirements for the operating staff. Reference 2 was submitted proposing a Certified Fuel Handler Training and Retraining Program for NRC approval.

This request also proposes to eliminate the Explosive Gas and Storage Tank Radioactivity Monitoring Program. A commitment to release all gases in the Radioactive Waste Storage System gas decay tanks prior to implementation of the requested License Amendment is contained in Attachment D.

FPC requests approval of this LAR by October 31, 2013, with a 30 day implementation period.

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

Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428

A001
NRR

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Licensing Supervisor, at (352) 563-4796.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 25, 2013.

Sincerely,

John Elnitsky, Vice President
Project Management and Construction

JE/scp

Attachments:
A. Description of Proposed License Amendment Request, Background, Justification for the Request, and Regulatory Analysis
B. Proposed Technical Specification Page Changes, Strikeout and Shadowed Text Format
C. Proposed Technical Specification Page Changes, Revision Bar Format
D. List of Regulatory Commitments

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

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #313, REVISION 0

ATTACHMENT A

DESCRIPTION OF PROPOSED LICENSE AMENDMENT REQUEST, BACKGROUND, JUSTIFICATION FOR THE REQUEST, AND REGULATORY ANALYSIS

DESCRIPTION OF PROPOSED LICENSE AMENDMENT REQUEST, BACKGROUND, JUSTIFICATION FOR THE REQUEST, AND REGULATORY ANALYSIS

1.0 Description of Proposed License Amendment Request

Pursuant to 10 CFR 50.90, Florida Power Corporation (FPC) proposes to amend the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS). This License Amendment Request (LAR) proposes to revise and remove certain requirements from the Section 5, Administrative Controls, portions of the ITS that are no longer applicable to CR-3 in the permanently defueled condition.

2.0 Background

CR-3 has been shutdown since September 26, 2009, when the plant entered the Cycle 16 refueling outage. In the process of creating a construction opening for replacement of steam generators during that outage, a delamination of the outer concrete shell of the containment was discovered. The construction opening and adjacent concrete shell of the containment was repaired during 2010 and 2011. During tensioning of the containment prestressing tendons following the concrete repair, delaminations occurred in two other sections of the containment shell. In consideration of performing a second repair of the containment shell, all fuel was removed from the reactor vessel and placed in storage in the Spent Fuel Pools as of May 28, 2011. On February 5, 2013, Progress Energy Florida, a subsidiary of Duke Energy, announced that CR-3 would be retired. The NRC has acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessel, and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

3.0 Justification For The Request

The decay heat load in the spent fuel pools is low since freshly irradiated fuel was last added to the pools three and one half years ago. Therefore, due to the low decay heat load, significant time is available to respond to loss of cooling or loss of inventory events. The following generic conclusions from NUREG-1738 and NUREG-1275 demonstrate that CR-3 is in a low risk condition that supports the ITS changes proposed in this LAR.

- Based on conservative calculation results presented in NUREG-1738, "Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants," (Reference 1) the time to boil off Spent Fuel Pool inventory down to three feet above the top of the fuel would be approximately 14.5 days with no action taken to restore cooling or inventory. Even this reduced inventory condition would not result in any significant effect to members of the public. In addition, this long decay time has resulted in essentially depleting the radioactive iodine nuclides from the pool inventory, removing the most significant contributor to offsite consequences.

- Spent fuel cooling is being provided by the systems normally used during plant operation and is as described in the CR-3 Final Safety Analysis Report. These systems are typical for a pressurized water reactor plant. Therefore, the conclusions of NUREG-1275, Volume 12, "Operating Experience Feedback Report - Assessment of Spent Fuel

Cooling," are applicable to CR-3. The report concludes that based on 12 years of operating experience, the staff determined that loss of spent fuel pool coolant inventory has occurred at a rate of about 1 event per 100 reactor years and that none of these events resulted in a water level less than 20 feet above the fuel. It also concludes that loss of cooling with a temperature increase of 20°F has occurred at a rate of approximately 3 events per 1000 reactor years.

The changes proposed herein are consistent with the operating experience and changes made by other permanently defueled plants and the low risk conditions that exist at CR-3. The following table identifies each section that is being changed, the proposed changes, and the basis for the changes:

CR-3 Specification	**Proposed Change and Basis**
5.1 Responsibility	
5.1.1	This section defines the responsible position for overall unit operation and for approval of each proposed test, experiment or modification to systems or equipment that affect stored nuclear fuel. The position title is changed from Plant General Manager to Plant Manager; and the scope of the position is changed from the effect on nuclear safety to the effect on stored nuclear fuel.
5.1.2	This section identifies the responsibilities for the control room command function associated with Modes of plant operation, and is based on personnel positions and qualifications for an operating plant. It identifies the need for a delegation of authority for command in an operating plant when the principal assignee leaves the control room. This section is being changed to eliminate the MODE dependency for this function and personnel qualifications associated with an operating plant. The proposed change establishes the Shift Supervisor as having command of the shift. Delegation of command is unnecessary for CR-3 where all fuel is in the spent fuel pools, and no fuel has been in a critical core for over three and a half years. Any event involving loss of pool cooling would evolve slowly enough that no immediate control room response would be required to protect the health and safety of the public or station personnel.
5.2 Organization	
5.2.1 Onsite and Offsite Organizations 5.2.1.a	The introduction to this section identifies that organizational positions are established that are responsible for the safety of the nuclear plant. This is changed to require that positions be established that are

	responsible for the safe handling and storage of nuclear fuel. This change removes the implication that CR-3 can return to operation.
5.2.1.b	This section identifies the organizational position responsible for overall nuclear plant safety, for the safe operation of the plant, and for control of activities necessary for the safe operation and maintenance of the plant. To reflect the reduced safety concerns from an operating plant to a permanently defueled plant, the responsibility for overall nuclear safety is changed to the overall responsibility for safe handling and storage of nuclear fuel. The assignment of this responsibility is changed from the Vice President – Crystal River Nuclear Plant to the Decommissioning Director. The responsibility to control those onsite activities necessary for safe operation and maintenance of the plant is changed to control those onsite activities necessary for safe handling and storage of nuclear fuel and is changed from the Vice President – Crystal River Nuclear Plant to the Plant Manager.
5.2.1.c	This paragraph addresses the requirement for organizational independence of the operations, health physics and quality assurance personnel from operating pressures. This is changed to replace "operating staff" with "Certified Fuel Handlers" and to replace "their independence from operating pressures" to "their ability to perform their assigned functions." These changes reflect the changed function of the previous operating staff to a focus on safe handling and storage of nuclear fuel, and to remove the implication that CR-3 can return to operation.
5.2.2 Unit Staff 5.2.2.a	This paragraph addresses that one auxiliary nuclear operator must be assigned to the operating shift whenever fuel is in the reactor. Since this can never occur again at CR-3, the minimum requirement is changed to a minimum crew compliment of one Shift Supervisor and one Non-certified Operator. This reflects the reduced demand on the operating crew to maintain the safety of fuel stored in the fuel pools. The Certified Fuel Handler will be the Shift Supervisor in accordance with new paragraph 5.2.2.e. In this position, he will retain command and control responsibility for operational decisions and will be responsible for the functions required for event reporting and emergency response.
5.2.2.b	This paragraph addresses the conditions under which the minimum shift compliment may be reduced. It contains a reference to 10 CFR 50.54(m) which establishes the minimum requirements for a licensed operating staff for facility operation. This reference is removed since CR-3 will not return to operation in

	the future, and the requirement for licensed operating personnel will no longer be required to protect public health and safety.
5.2.2.c	This paragraph establishes the requirement for one licensed Reactor Operator to be in the control room when fuel is in the reactor, and for one Senior Reactor Operator to be in the control room during operating Modes 1 – 4. This paragraph is changed to reflect the requirement for having one qualified watch stander (either a Non-certified Operator or Certified Fuel Handler) in the control room when fuel is stored in the spent fuel pools. This reflects the reduced requirement for control room personnel training and qualification for a plant authorized for nuclear fuel storage only. CR-3 has submitted a Certified Fuel Handler Training and Retraining Program for NRC approval. The training and qualification for the Non-certified Operator will be determined in accordance with the systems approach to training (SAT) as defined in 10 CFR 55.4. This process ensures that the Non-certified Operator will be qualified to perform the functions necessary to monitor and ensure safe fuel storage is maintained. The SAT process requires (1) systematic analysis of the jobs to be performed, (2) learning objectives derived from the analysis which describe desired performance after training, (3) training design and implementation based on the learning objectives, (4) evaluation of trainee mastery of the objectives during training, and (5) evaluation and revision of the training based on the performance of trained personnel in the job setting. For any conditions, incidents, or events that occur when the Non-certified Operator is in the control room alone and are not within the scope of qualifications that are possessed by the Non-certified Operator, the Certified Fuel Handler will immediately be contacted for direction by phone, radio, and/or plant page system. This philosophy is deemed acceptable because the necessity to render immediate actions to protect the health and safety of the public is not challenged. A conservative engineering calculation indicates upon a total loss of spent fuel pool cooling the temperature in the spent fuel pool will take approximately four days to reach 200° F.
5.2.2.d	This paragraph established the requirement for a person qualified in Radiation Protection procedures to be onsite when fuel is in the reactor. This paragraph is deleted since CR-3 is no longer authorized to have fuel in the reactor. Deletion of this paragraph recognizes the expanded response time available for radiation protection staff, in the event of a loss of cooling or inventory in the spent fuel pools, as compared to power operation considering the stored fuel exposure

	history.
5.2.2.d (New)	A new paragraph is added to establish the requirement for having oversight of fuel handling operations performed by a Certified Fuel Handler.
5.2.2.e (New)	A new paragraph is added to establish that the Shift Supervisor must be a Certified Fuel Handler. In the permanently defueled plant, the Certified Fuel Handler is the senior position on the operating crew. It is not necessary for the Shift Supervisor to hold a Senior Reactor Operator license if the plant cannot operate to generate power.
5.3 Unit Staff Qualifications	
5.3.1	This paragraph establishes that the unit staff must meet or exceed the minimum qualifications of ANSI N18.1, 1971 and for the Radiation Protection Manager to meet the qualifications of NRC Regulatory Guide 1.8, September 1975. The paragraph also establishes the requirements for the Shift Technical Advisor. This paragraph is changed to remove the requirements for the Shift Technical Advisor since that position is only required for a plant authorized for power operations.
5.3.2 (New)	This new paragraph is added to identify that responsibility for training and retraining of Certified Fuel Handlers is assigned to the Plant Manager.
Sections 5.4 and 5.5 are not currently used.	
5.6 Procedures. Programs, and Manuals	
5.6.1 Procedures	
5.6.1.1 Scope 5.6.1.1.a	This section states the requirement for procedures to be established, implemented and maintained covering various plant activities. Subparagraph (a) establishes a requirement to have applicable procedures recommended in NRC Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. This requirement is changed to reduce the scope to procedures applicable to the safe storage of nuclear fuel recommended in NRC Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. This recognizes the reduced requirements associated with protection of the stored nuclear fuel.
5.6.2 Programs and Manuals	
5.6.2.1 Not Used	

5.6.2.2 Not Used	
5.6.2.3 Offsite Dose Calculation Manual (ODCM)	In 5.6.2.3.2, the authority for approval of changes to the ODCM is changed from the Plant General Manager to the Plant Manager, consistent with the position title change in 5.1.1.
5.6.2.4 Primary Coolant Sources Outside Containment	This program was established to minimize leakage from portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The program is being eliminated since these conditions can no longer exist for a permanently defueled plant
5.6.2.5 Component Cyclic or Transient Limit	This program provided controls to track cyclic and transient occurrences to ensure that components were maintained within their design limits. The program is being eliminated since serious transient or accident conditions can no longer exist for a permanently defueled plant, and the monitored components are not required to assure spent fuel cooling.
5.6.2.6 Not Used	
5.6.2.7 Not Used	
5.6.2.8 Inservice Inspection Program	This program established the controls for periodic inspection of ASME Code Class 1, 2, 3, MC and CC components including applicable supports in accordance with ASME Section XI. The Preface to ASME Section XI states: "The rules of this section constitute requirements to maintain the nuclear power plant and to return the plant to service, following plant outages, in a safe and expeditious manner. The rules require a mandatory program of examinations, testing, and inspections to evidence adequate safety and to manage aging and deterioration effects." This program is no longer required since CR-3 is permanently defueled and cannot operate. Therefore, the ASME Section XI systems and components will not be subjected to the temperature and pressure effects that the Inservice Inspection Program was in place to protect against.
5.6.2.9 Inservice Testing Program	No Changes
5.6.2.10 Steam Generator (OTSG) Program	The Steam Generator Program established and implemented practices to ensure that OTSG tube integrity was maintained. This program is no longer required since CR-3 is permanently

	defueled and cannot operate. Therefore, the steam generator tubes will not be subjected to the temperature and pressure effects that the Steam Generator Program was in place to protect against.
5.6.2.11 Secondary Water Chemistry Program	This program provided controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. This program is no longer required since CR-3 is permanently defueled and cannot operate. Secondary systems are drained.
5.6.2.12 Ventilation Filter Testing Program (VFTP)	No Changes
5.6.2.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program	This program provided controls for potentially explosive gas mixtures contained in the Radioactive Waste Disposal (WD) System, and the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system. In Attachment D to this LAR, a Regulatory Commitment is made to vent and remove from service the Radioactive Waste System gas decay tanks by August 30, 2013. Based on that commitment, this program is being eliminated.
5.6.2.14 Diesel Fuel Oil Testing Program	No Changes
5.6.2.15 Not Used	
5.6.2.16 Safety Function Determination Program (SFDP)	No Changes
5.6.2.17 Technical Specification (TS) Bases Control Program	No Changes
5.6.2.18 Core Operating Limits Report (COLR)	This program established that core operating limits be established prior to each reload cycle. This program is being eliminated since no reactor core can be reloaded into the CR-3 reactor.
5.6.2.19 Reactor Coolant System (RCS) Pressure And Temperature Limits Report (PTLR)	This program ensured that RCS pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, be established and documented in the PTLR. Per NRC Regulatory Guide 1.184, this program is being eliminated. The reactor coolant piping has been drained and is not subject to pressurization. The reactor vessel contains water, but is vented and not subject to pressurization.
5.6.2.20 Containment Leakage Rate Testing	This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50,

Program	Appendix J, Option B, as modified by approved exemptions. Per NRC Regulatory Guide 1.184, this program is being eliminated.
5.6.2.21 Control Complex Habitability Envelope Integrity Program	No Changes
5.7 Reporting Requirements	
5.7.1 Routine Reports	No Changes
5.7.2 Special Reports	The Special Reports section is being eliminated. The Special Reports that currently exist are associated with plant conditions that cannot exist in a permanently defueled plant or for programs that are being eliminated.
5.8 High Radiation Area	
5.8.2	The first paragraph contains the requirements for control of keys to areas with radiation levels $\geq$ 1000 mrem/hr at 30 cm. It identifies that one of the personnel responsible for control of the keys is the Control Room Supervisor. This is being changed to the Shift Supervisor consistent with 5.1.2.

4.0 Regulatory Analysis

4.1 No Significant Hazards Consideration Determination

License Amendment Request (LAR) #313, Revision 0, seeks NRC approval to change certain requirements from the Section 5, Administrative Controls, portions of the Crystal River Unit 3(CR-3) Improved Technical Specifications (ITS) that are no longer applicable as CR-3 is in a permanently defueled condition. Florida Power Corporation (FPC) has evaluated whether or not the proposed changes would result in a significant hazards consideration by application of the three standards set forth in 10 CFR 50.92(c). No significant hazards will exist if the proposed changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. Involve a significant reduction in a margin of safety.

The following table provides the evaluation of each of the proposed changes that provides the basis for the determination that no significant hazards consideration is involved.

SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Identification and Description of Change	Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?	Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?	Does the proposed change involve a significant reduction in a margin of safety?
5.1.1 This section defines the responsible position for overall unit operation and for approval of each proposed test, experiment or modification to systems or equipment that affect stored nuclear fuel and fuel handling. The responsible position title is changed from the Plant General Manager to the Plant Manager.	No. The change reflects that the remaining credible accident is a fuel handling accident or loss of spent fuel cooling. The change in the position title of the responsible person is administrative and cannot increase the probability or consequences of a fuel handling accident.	No. This change reflects an organizational change to transition from an operating plant to a permanently defueled plant. Such an administrative change cannot create a new or different kind of accident.	No. The position title proposed here does not involve any physical plant limits or parameters and therefore cannot affect any margin of safety.
5.1.2 This section identifies the responsibilities for the control room command function associated with Modes of plant operation, and is based on personnel positions and qualifications for an operating plant. It identifies the need for a delegation of authority for command in an operating plant when the principal assignee leaves the control room. This section is being changed to eliminate the MODE dependency for this function and personnel qualifications associated with an	No. This is a change to the requirements for control room staffing. In a permanently defueled plant, the fuel handling accident is the only credible accident previously evaluated. This action cannot increase the probability or consequences of a fuel handling accident.	No. The changes proposed here for control room staffing cannot create a new or different kind of accident since they do not change the function of any plant structures, systems, or components.	No. The changes proposed here for control room staffing do not directly involve any limits or parameters and therefore cannot affect any margin of safety.

operating plant. The proposed change establishes the Shift Supervisor as having command of the shift.			
5.2.1.a The introduction to this section identifies that organizational positions are established that are responsible for the safety of the nuclear plant. This is changed to require that positions be established that are responsible for the safe storage and handling of nuclear fuel. This change removes the implication that CR-3 can return to operation.	No. This change in the description of the functional responsibility of organizational positions places emphasis on the safe storage and handling of nuclear fuel. This focus on their principal responsibility cannot increase the probability or consequences of a fuel handling accident.	No. This change in the description of the functional responsibility of organizational positions cannot create a new or different kind of accident since they do not change the function of any plant structures, systems, or components.	No. This change does not directly involve any physical limits or parameters and therefore cannot affect any margin of safety.
5.1.2.b This section identifies the organizational position responsible for overall nuclear plant safety, for the safe operation of the plant, and for control of activities necessary for the safe operation and maintenance of the plant. This section is being changed to recognize that the safety concerns for a permanently defueled plant are for the safe storage and handling of nuclear fuel. It changes responsibility for overall safety for storage and handling of nuclear fuel to the Decommissioning Director. It changes responsibility for control over onsite activities necessary for safe handling and storage of nuclear fuel to the Plant Manager.	No. This change in the description of the functional responsibility of organizational positions places emphasis on the safe handling and storage of nuclear fuel. This focus on their principal responsibility cannot increase the probability or consequences of a fuel handling accident.	No. This change in the description of the functional responsibility of organizational positions cannot create a new or different kind of accident since they do not change the function of any plant structures, systems, or components.	No. This change does not directly involve any physical limits or parameters and therefore cannot affect any margin of safety.
5.2.1.c This paragraph addresses the requirement for organizational independence of	No. This change continues to ensure that personnel in	No. This change does not introduce any	No. This change does not directly involve any

the operations, health physics and quality assurance personnel from operating pressures. This is changed to replace "operating staff" with "Certified Fuel Handlers, " and to replace "their independence from operating pressures" to "their ability to perform their assigned functions. "	specifically identified positions retain independence from organizational pressures and will not increase the probability or occurrence of a fuel handling accident.	changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	limits or parameters and therefore cannot affect any margin of safety.
5.2.2.a This paragraph addresses that one auxiliary nuclear operator must be assigned to the operating shift whenever fuel is in the reactor. Since this can never occur again at CR-3, the minimum requirement is changed to a minimum crew compliment of one Shift Supervisor and one Non-certified Operator.	No. This change, in conjunction with new paragraph 5.2.2.e, continues to ensure that personnel trained and qualified for the safe handling and storage of nuclear fuel are onsite. This cannot increase the probability or consequences of a fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.2.2.b This paragraph addresses the conditions under which the minimum shift compliment may be reduced. It contains a reference to 10 CFR 50.54(m) which establishes the minimum requirements for a licensed operating staff for facility operation. This reference is removed since CR-3 will not return to operation in the future, and the requirement for licensed operating personnel will no longer be required to protect public health and safety.	No. This change continues to ensure that the minimum shift compliment of qualified personnel will not be decreased for more than a limited period. It removes the qualification requirements for personnel who are capable of responding to operating plant transients and accidents. This does not involve an increase in the	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

	probability or consequences of a fuel handling accident.		
5.2.2.c This paragraph establishes the requirement for one licensed Reactor Operator to be in the control room when fuel is in the reactor and for one Senior Reactor Operator to be in the control room during operating Modes 1 – 4. The change establishes the requirements for either a Non-certified operator or Certified Fuel Handler to be in the control room when fuel is stored in the pools.	No. This change continues to ensure that personnel trained and qualified for the handling and storage of nuclear fuel man the control room. This cannot increase the probability or consequences of a fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.2.2.d This paragraph established the requirement for a person qualified in Radiation Protection procedures to be onsite when fuel is in the reactor. This paragraph is deleted since CR-3 is no longer authorized to have fuel in the reactor.	No. This is an administrative change that cannot affect the probability of a fuel handling accident. The consequences of a fuel handling accident are governed by the characteristics of the fuel element and are not affected by the presence or absence of radiation protection trained personnel.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.2.2.d (New) A new paragraph is added to establish the requirement for having oversight of fuel handling operations to be performed by a Certified Fuel Handler.	No. Certified Fuel Handlers are specifically trained and qualified to safely handle irradiated	No. This change does not introduce any changes to the function of any plant structures,	No. This change does not directly involve any limits or parameters and therefore cannot affect

	fuel. Applying these qualifications to fuel movement ensures that the probability or consequences of a fuel handling accident are not increased.	systems, or components therefore it cannot create a new or different kind of accident.	any margin of safety.
5.2.2.e (New) A new paragraph is added to establish that the Shift Supervisor must be a Certified Fuel Handler. In the permanently defueled plant, the Certified Fuel Handler is the senior position on the operating crew. It is not necessary for the Shift Supervisor to hold a Senior Reactor Operator license if the plant cannot operate to generate power.	No. Certified Fuel Handlers are specifically trained and qualified to safely handle irradiated fuel. Applying these qualifications to the supervision of fuel movement ensures that the probability or consequences of a fuel handling accident are not increased.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.3.1 This paragraph is changed to remove the requirements for the Shift Technical Advisor since that position is only required for a plant authorized for power operations. The paragraph retains the previous requirements for the personnel filling unit staff positions meet or exceed the minimum qualifications of ANSI N18.1, 1971, and the Radiation Protection Manager meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.	No. The Shift Technical Advisor position was established to assist the control room operating personnel to diagnose the cause and advise on the response to operating transients and accidents. The absence of a staff member with those qualifications does not change the probability or consequences of a fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any physical equipment limits or parameters and therefore cannot affect any margin of safety.

5.3.2 This new paragraph is added to identify that responsibility for the training and retraining of Certified Fuel Handlers is assigned to the Plant Manager.	No. This section recognizes the importance of establishing and maintaining Certified Fuel Handler qualifications and assigns a manager responsibility for this program. Training and retraining Certified Fuel Handlers specifically trained to safely handle nuclear fuel will not increase the probability or consequences of a fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any physical equipment limits or parameters and therefore cannot affect any margin of safety.
5.6.1.1.a This section states the requirement for procedures to be established, implemented and maintained covering various plant activities. The scope is reduced to procedures applicable to the safe handling and storage of nuclear fuel.	No. The procedures necessary for the safe handling of nuclear fuel are included in the group of procedures applicable to the safe storage of nuclear fuel. With these procedures in effect for fuel handling, the probability or consequences of a fuel handling accident will not be increased.	No. The applicable procedures for the safe storage of nuclear fuel will direct the correct use of fuel handling equipment. These procedures are currently in place and have been used effectively for the safe handling of fuel. These procedures will not direct the use of plant structures, systems, or components in a different manner, therefore, they cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

5.6.2.3 In this section, the authority for approval of changes to the Offsite Dose Calculation Manual (ODCM) is changed from the Plant General Manager to the Plant Manager consistent with the position title change in 5.1.1.	No. This is a change to the requirements for the position responsible for approving ODCM changes. In a permanently defueled plant, the fuel handling accident is the only credible accident previously evaluated. This action cannot increase the probability or consequences of a fuel handling accident.	No. The change proposed here, identifying a different position responsible for ODCM change approval, cannot create a new or different kind of accident since this does not change the function of any plant structures, systems, or components.	No. The changes proposed here for ODCM approval do not directly involve any limits or parameters for operating systems and therefore cannot affect any margin of safety.
5.6.2.4 Primary Coolant Sources Outside Containment This program was established to minimize leakage from portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The program is being eliminated.	No. The fuel handling accident is the only credible accident for a permanently defueled plant. This change eliminates an inspection program that is no longer necessary to limit the consequences of operating transients and accidents. This change cannot increase the probability or consequences of the fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.6.2.5 Component Cyclic or Transient Limit This program provided controls to track cyclic and transient occurrences to ensure that components were maintained within their	No. Eliminating an administrative event tracking program cannot increase the probability or consequences of a fuel handling accident.	No. Eliminating an administrative event tracking program cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

design limits. This program is being eliminated.			
5.6.2.8 Inservice Inspection Program This program required periodic inspections, examinations, and tests of plant pressure boundary components to ensure their continued integrity for power operation. This program is being eliminated.	No. The Inservice Inspection Program does not apply to nuclear fuel or fuel handling equipment. Therefore eliminating this program cannot increase the probability or occurrence of a fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. For an operating plant the Inservice Inspection Program provided confidence that plant systems that were either a potential source of an accident or transient or served to mitigate events continued to meet their physical design requirements. For a permanently shutdown plant, no transient or accident can occur, so ending this inspection program cannot affect any margin of safety.
5.6.2.10 Steam Generator (OTSG) Program The Steam Generator Program established and implemented practices to ensure that OTSG tube integrity was maintained. This program is being eliminated.	No. The condition of the steam generator tubes inside the containment has no effect on fuel handing in the auxiliary building within the spent fuel pools. Therefore, eliminating the program cannot increase the probability or occurrence of a fuel handling accident.	No. The CR-3 steam generators will remain out of service until removed from the plant. In this state, the condition of the steam generator tubes is immaterial and cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.6.2.11 Secondary Water Chemistry Program	No. The secondary piping	No. This change does	No. The components this

This program provided controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. This program is being eliminated.	systems do not interconnect with the fuel cooling or fuel handling systems. Therefore, eliminating the Secondary Water Chemistry Program cannot increase the probability or occurrence of a fuel handling accident.	not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	program was intended to protect will no longer function for power production. Therefore, eliminating this program cannot affect any margin of safety.
5.6.2.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program This program provided controls for potentially explosive gas mixtures contained in the Radioactive Waste Disposal (WD) System, and the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system. This program is being eliminated.	No. This program is required for an operating plant where hydrogen and radioactive gases are created and must be controlled. Controlled release of any gases currently in the tanks, in accordance with existing procedures, will ensure there will be no hazard to public heath and safety. Therefore, elimination of this program cannot increase the probability or consequences of a fuel handling accident.	No. This program is required for an operating plant where hydrogen and radioactive gases are created and must be controlled. Controlled release of any gases currently in the tanks, in accordance with existing procedures, will ensure there will be no hazard to public heath and safety. Therefore, elimination of this program cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.6.2.18 Core Operating Limits Report (COLR) This program established that core operating limits be established prior to each reload cycle. This program is being eliminated.	No. This program for controlling the design and operation of the reactor core has no bearing on fuel storage after fuel has been moved into the spent fuel pools. Therefore,	No. Since CR-3 can never load a core into the reactor again, eliminating this control program cannot create a new or different kind of accident.	No. Since CR-3 can never load a core into the reactor again, eliminating this control program cannot affect any margin of safety.

	eliminating this program cannot increase the probability or occurrence of a fuel handling accident.		
5.6.2.19 Reactor Coolant System (RCS) Pressure And Temperature Limits Report (PTLR) This program ensured that RCS pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, be established and documented in the PTLR. This program is being eliminated.	No. This program contains no actions or limits that affect the storage or handling of nuclear fuel. Therefore, eliminating this program cannot increase the probability or occurrence of a fuel handling accident.	No. This report is no longer needed since the reactor coolant system is not subject to pressurization and the reactor contains no fuel. Therefore, eliminating this control program cannot create a new or different kind of accident.	No. The limits established in this report do not apply to nuclear fuel stored in the spent fuel pools. Therefore, eliminating this program cannot affect any margin of safety.
5.6.2.20 Containment Leakage Rate Testing Program This program was established to implement the leakage rate testing of the containment. This program is being eliminated in accordance with Regulatory Guide 1.184.	No. Since fuel can never be returned to the CR-3 containment, ending containment leakage rate testing cannot increase the probability or occurrence of a fuel handling accident.	No. This change does not introduce any changes to the function of any plant structures, systems, or components therefore it cannot create a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
5.7.2 Special Reports This section is being eliminated.	No. Eliminating reporting requirements for programs that are no longer required or conditions that cannot exist in a permanently defueled plant cannot increase the probability or occurrence of a fuel handling accident.	No. Eliminating reporting requirements that are no longer required cannot create a new or different kind of accident.	No. Eliminating reporting requirements that are no longer required cannot affect any margin of safety.

5.8.2 High Radiation Area Controls Changes one of the personnel responsible for locked high radiation area key control from the Control Room Supervisor to the Shift Supervisor.	No. This is a change to the requirements for the position title responsible for key control. In a permanently defueled plant, the fuel handling accident is the only credible accident previously evaluated. This action cannot increase the probability or consequences of a fuel handling accident.	No. The change proposed here, identifying a different position title responsible for key control, cannot create a new or different kind of accident since they do not change the function of any plant structures, systems, or components.	No. The changes proposed here for key control do not directly involve any limits or parameters and therefore cannot affect any margin of safety.

4.2 Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if the amendment changes a requirement with respect to use of a facility component within the restricted area provided that (i) the amendment involves no significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this LAR and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The following is the basis for this determination:

(i) The proposed license amendment does not involve a significant hazards consideration, as described in the significant hazards evaluation.

(ii) As discussed in the Justification for the Request and the No Significant Hazards Consideration, this change does not result in a significant change or significant increase in the release associated with any Design Basis Accident. There will be no significant change in the types or a significant increase in the amounts of any effluents released offsite during normal operation. There will be no significant change in the types or increase in the amounts of any effluents that may be released offsite and does not involve irreversible environmental consequences beyond those already associated with the CR-3 Final Environmental Statement.

(iii) The proposed LAR does not result in a significant increase to the individual or cumulative occupational radiation exposure because this is a change to plant equipment that does not interface with radiologically contaminated systems and does not require operator or other actions that could increase occupational radiation exposure. Therefore, the proposed LAR does not result in a significant increase to the individual or cumulative occupational radiation exposure.

4.3 Applicable Regulatory Requirements/Criteria

10 CFR 50.82(a)(1) requires that when a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of § 50.4(b)(8), and once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of § 50.4(b)(9). CR-3 submitted the required certifications by letter dated February 20, 2013. The NRC acknowledged receipt of the required certifications by letter dated March 13, 2013.

10 CFR 50.36 establishes the requirements for Technical Specifications. 50.36(c)(5), *Administrative Controls,* identifies that an Administrative Controls section shall be included in the Technical Specifications and shall include provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. This LAR is proposing changes to the Administrative Controls section consistent with the decommissioning status of the plant. This LAR applies the principles identified in 50.36(c)(6), *Decommissioning*, for a facility which has submitted certification required by 50.82(a)(1) and proposes changes to the Administrative Controls appropriate for the CR-3 permanently defueled condition. As 50.36(c)(6) states, this type of change should be considered on a case-by-case basis.

10 CFR 50.54(m) establishes the requirements for having Reactor Operators and Senior Reactor Operators licensed in accordance with Part 55 based on plant conditions. Based on the permanent cessation of operation for CR-3, the requirements of this section no longer apply and it is permissible to remove those positions from the Technical Specifications.

10 CFR 50.55a establishes that each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section. 50.55a(g)(1) requires that a plant whose construction permit was issued before January 1, 1971 must meet the requirements of paragraph (g)(4). The construction permit for CR-3 was issued September 25, 1968. 50.55a(g)(4) requires implementation of ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," throughout its service life. Since CR-3 has permanently ceased operation, its service life has ended and the CR-3 Improved Technical Specification (ITS) Program for Inservice Inspection can be eliminated from the ITS.

5.0 References

1. NUREG-1738, "Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants," February 2001 (ML010430066)
2. NUREG--1275, Volume 12, "Operating Experience Feedback Report - Assessment of Spent Fuel Cooling," February 1997 (ML010670175)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #313, REVISION 0

ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES, STRIKEOUT AND SHADOWED TEXT FORMAT

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Plant ~~General~~ Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Plant ~~General~~ Manager or his designee shall approve, prior to implementation, each proposed test, experiment or modifications to systems or equipment that affect nuclear safety.

5.1.2 ~~The Control Room Supervisor shall be responsible for the control room command function. During any absence of the Control Room Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Control Room Supervisor from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.~~
The Shift Supervisor shall be responsible for the shift command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions responsible for activities affecting ~~safety of the nuclear power plant.~~ the safe handling and storage ~~and handling~~ of nuclear fuel.

a. Lines of authority, responsibility, and communications shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These shall be documented in the FSAR;

b. The ~~Vice President - Crystal River Nuclear Plant~~ Decommissioning Director shall have ~~corporate responsibility for~~ overall responsibility for the safe handling and storage of nuclear fuel ~~plant nuclear safety~~ and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure the safe handling and storage of nuclear fuel~~safety~~. The ~~Vice President -Crystal River Nuclear~~ Plant Manager shall be responsible ~~for the overall safe operation of the plant and shall have~~ to control ~~over~~ those onsite activities necessary for the safe handling and storage of nuclear fuel~~operation and maintenance of the plant~~; and

c. The individuals who train the ~~operating staff~~ Certified Fuel Handlers, carry out health physics or perform quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure ~~their independence from operating pressures~~ their ability to perform their assigned functions.

5.2.2 Unit Staff

The unit staff organization shall include the following:

~~a. One auxiliary nuclear operator shall be assigned to the operating shift any time there is fuel in the reactor and~~

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

~~an additional auxiliary nuclear operator shall be assigned in MODES 1, 2, 3 and 4.~~

a. Each duty shift shall be composed of at least one Shift Supervisor and one Non-certified Operator.

b. Shift crew composition may be less than the minimum requirement of ~~10 CFR 50.54(m)(2)(i) and~~ 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

~~c. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

c. At least one person qualified to stand watch in the control room (Non-certified Operator or Certified Fuel Handler) shall be present in the control room when nuclear fuel is stored in the spent fuel pools.

~~d. An individual qualified in Radiation Protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.~~

d. Oversight of fuel handling operations shall be provided by a Certified Fuel Handler.

e. The Shift Supervisor shall be a Certified Fuel Handler.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1, 1971 for comparable positions, except for the Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~, and the Shift Technical Advisor who shall have a Bachelor's degree, or the equivalent, in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant transients and accidents~~.

5.3.2 A training and retraining program for the Certified Fuel Handler positions shall be maintained under the direction of the Plant Manager.

5.0 ADMINISTRATIVE CONTROLS

5.4 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.5 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.6 Procedures, Programs, and Manuals

5.6.1 Procedures

5.6.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The ~~applicable~~ procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. Quality assurance for effluent and environmental monitoring;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.6.2.

5.6.2 Programs and Manuals

The following programs shall be established, implemented, and maintained. Programs and Manuals may be titled as Reports.

5.6.2.1 Not Used

5.6.2.2 Not Used

5.6.2.3 Offsite Dose Calculation Manual (ODCM):

This Manual contains offsite dose calculation methodologies, the radioactive effluent controls program, and radiological environmental monitoring activities. The ODCM shall contain:

1. The methodologies and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents;

2. The methodologies and parameters used in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints;

3. The controls for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable in accordance with 10 CFR 50.36a. These include:

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.3 ODCM (continued)

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values of 10 CFR 20.1001 - 20.2401, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   1. For noble gases: Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.3 ODCM (continued)

2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Licensee Initiated Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the on-site review function and the approval of the Plant ~~General~~ Manager; and

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.3 ODCM (continued)

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date, (e.g., month/year) the change was implemented.

5.6.2.4 ~~Primary Coolant Sources Outside Containment~~ Not Used

~~This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Low Pressure Injection, Reactor Building Spray and Makeup and Purification. The program shall include the following:~~

~~a. Preventive maintenance and periodic visual inspection requirements; and~~

~~b. Integrated leak test requirements for each system at refueling cycle intervals or less.~~

5.6.2.5 ~~Component Cyclic or Transient Limit~~ Not Used

~~This program provides controls to track the FSAR Table 4.8, cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

5.6.2.6 Not Used

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.7 Not Used

5.6.2.8 ~~Inservice Inspection Program~~ Not Used

~~This program provides controls for inservice inspection of ASME Code Class 1, 2, 3, MC and CC components, including applicable supports. The program shall include the following:~~

~~a. Provisions that inservice inspection of ASME Code Class 1, 2, 3, MC and CC components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a;~~

~~b. The provisions of SR 3.0.2 are applicable to the frequencies for performing inservice inspection activities;~~

~~c. Inservice inspection of each reactor coolant pump flywheel shall be performed at least once every twenty years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination for exposed surfaces of the disassembled flywheels. The recommendations delineated in Regulatory Guide 1.14, Positions 3, 4, and 5 of Section C.4.b shall apply.~~

~~d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.~~

(continued)

5.6 Procedures, Programs and Manuals (continued)

5.6.2.9 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

b. Testing frequencies specified in the ASME OM Code and applicable Addenda;

c. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as two years or less in the Inservice Testing Program for performing inservice testing activities;

d. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.10 ~~Steam Generator (OTSG) Program~~ Not Used

~~A Steam Generator Program shall be established and implemented to ensure that OTSG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:~~

~~a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an OTSG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the OTSG tubes are inspected or plugged to confirm that the performance criteria are being met.~~

~~b. Performance criteria for OTSG tube integrity. OTSG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.~~

~~1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~(continued)~~

5.6 Procedures, Programs and Manuals

5.6.2.10 ~~OTSG Program (continued)~~

~~2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than an OTSG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all OTSGs and leakage rate for an individual OTSG. Leakage is not to exceed one gallon per minute per OTSG.~~

~~3. The operational LEAKAGE performance criterion is specified in LCO 3.4.12, "RCS Operational LEAKAGE."~~

~~c. Provisions for OTSG tube repair criteria. A tube found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.~~

~~d. Provisions for OTSG tube inspections. Periodic OTSG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that OTSG tube integrity is maintained until the next OTSG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

~~1. Inspect 100% of the tubes in each OTSG during the first refueling outage following OTSG replacement.~~

~~2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the OTSGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint~~

~~(continued)~~

5.6 Procedures, Programs and Manuals

5.6.2.10 ~~OTSG Program (continued)~~

~~of the~~ period ~~and the remaining 50% by the refueling outage nearest the end of the period. No OTSG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

~~3. If crack indications are found in any OTSG tube, then the next inspection for each OTSG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.~~

~~e. Provisions for monitoring operational primary to secondary LEAKAGE.~~

~~(continued)~~

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5.6 Procedures, Programs and Manuals

5.6.2.11 ~~Secondary Water Chemistry Program~~ Not Used

~~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~~ Not Used

~~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 ≥ 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

5.6.2.16 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable); or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.16 SFDP (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.6.2.17 Technical Specifications (TS) Bases Control Program

Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to the review process specified in the Quality Assurance Plan.

Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

Proposed changes that meet the criteria of Specification 5.6.2.17.a or Specification 5.6.2.17.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.

5.6.2.18 ~~CORE OPERATING LIMITS REPORT (COLR)~~ Not Used

~~a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:~~

~~SL 2.1.1.1 API Protective Limit~~
~~LCO 3.1.1 SHUTDOWN MARGIN~~
~~SR 3.1.7.1 API/RPI Position Indication Agreement~~
~~LCO 3.1.3 Moderator Temperature Coefficient (MTC)~~
~~LCO 3.2.1 Regulating Rod Insertion Limits~~
~~LCO 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits~~

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.18 ~~COLR (continued)~~

~~LCO 3.2.3 AXIAL POWER IMBALANCE Operating Limits~~
~~LCO 3.2.4 QUADRANT POWER TILT~~
~~LCO 3.2.5 Power Peaking Factors~~
~~LCO 3.3.1 Reactor Protection System (RPS) Instrumentation~~
~~SR 3.4.1.1 Reactor Coolant System Pressure DNB Limits~~
~~SR 3.4.1.2 Reactor Coolant System Temperature DNB Limits~~
~~SR 3.4.1.3 Reactor Coolant System Flow DNB Limits~~
~~LCO 3.9.1 Boron Concentration~~

~~b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC:~~

~~BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed) and License Amendment 144, SER dated June 25, 1992. The approved revision number for BAW-10179P-A shall be identified in the COLR.~~

~~c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~

~~d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.~~

5.6.2.19 ~~Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~ Not Used

~~a. Other Applicable ITS:~~

~~3.4.3 RCS P/T Limits~~
~~3.4.11 Low Temperature Overpressure Protection~~

~~b. RCS pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. The analytical methods used to determine the pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and approved by the NRC in BAW-10046A, Rev. 2, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," June 1986. The analytical method used to determine vessel fluence shall be those reviewed by the NRC and documented in BAW-2241P, May 1997. The analytical method used to determine LTOP limits shall be those previously reviewed by the NRC based on ASME Code Case N-514. The Materials Program is in accordance with BAW-1543A, "Integrated Reactor Vessel Surveillance Program."~~

(continued)

5.6 Procedures, Programs and Manuals

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

~~c. 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.~~

~~d. 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~~ Not Used

~~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, as modified by the following exception:~~

~~1. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2006.~~

~~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,~~ $\cancel{L_a}$~~, at~~ $\cancel{P_a}$~~, shall be 0.25% of primary containment air weight per day.~~

~~Leakage Rate acceptance criteria are:~~

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

~~2. Air lock testing acceptance criteria are:~~

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

~~b. For each door, leakage rate is~~ $\cancel{\leq 0.01\ L_a}$ ~~when tested at~~ $\cancel{\geq 8.0}$ ~~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 Control Complex Habitability Envelope Integrity Program shall be established and implemented to ensure that CCHE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CCHE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a challenge from smoke. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CCHE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements.

1. The definition of the CCHE and the CCHE boundary.
2. Requirements for maintaining the CCHE boundary in its design condition including configuration control and preventive maintenance.
3. Requirements for (i) determining the unfiltered air in-leakage past the CCHE boundary into the CCHE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CCHE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.
4. The Control Complex Habitability Envelope Integrity Program will be used to verify the integrity of the Control Complex boundary. Conditions that are identified to be adverse shall be trended and used as part of the 24 month assessment of the CCHE boundary.
5. The quantitative limits on unfiltered air in-leakage into the CCHE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air in-leakage measured by the testing described in paragraph 3. The unfiltered air in-leakage limit for radiological challenges is the in-leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air in-leakage limits for hazardous chemicals and smoke must ensure that exposure of CCHE occupants to these hazards will be within the assumptions in the licensing basis.
6. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CCHE habitability, determining CCHE unfiltered in-leakage as required by paragraph 3.

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5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

5.7.1.1 Reports required on an annual basis include:

a. Not Used

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM).

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

c. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted prior to May 1 of each year, and in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program, and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV B.1.

(continued)

5.7 Reporting Requirements

5.7.1.2 Not Used

5.7.2 ~~Special Reports~~Not Used

~~Special Reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.~~

~~The following Special Reports shall be submitted:~~

~~a. When a Special Report is required by Condition B or F of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.~~

~~b. Any abnormal degradation of the containment structure found during the inspection performed in accordance with ITS 5.6.2.8 shall be reported to the NRC within 30 days of the current surveillance completion. The abnormal degradation shall be defined as findings such as delamination of the dome concrete, widespread corrosion of the liner plate, corrosion of prestressing elements (wires, strands, bars) or anchorage components extending to more than two tendons and group tendons force trends not meeting the requirements of 10CFR50.55a(b)(2)(ix)(B). The report shall include the description of degradation, operability determination, root cause determination and the corrective actions.~~

~~c. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.6.2.10, Steam Generator (OTSG) Program. The report shall include:~~

~~1. The scope of inspections performed on each OTSG,~~

~~2. Active degradation mechanisms found,~~

~~3. Nondestructive examination techniques utilized for each degradation mechanism,~~

~~4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,~~

~~(continued)~~

5.7 Reporting Requirements

5.7.2 ~~Special Reports (continued)~~

~~5. Number of tubes plugged_during the inspection outage for each active degradation mechanism,~~

~~6. Total number and percentage of tubes plugged_to date,~~

~~7. The results of condition monitoring, including the results of tube pulls and in-situ testing.~~

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), alternative methods are used to control access to high radiation areas. Each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation (measured at 30 cm) is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance.

5.8.2 In addition to the requirements of Specification 5.8.1, areas with radiation levels ≥ 1000 mrem/hr at 30 cm shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the ~~Control Room~~ Shift Supervisor or health physics supervision. Doors shall remain locked except during periods of access by personnel.

Direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

(continued)

5.8 High Radiation Area (continued)

5.8.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that are not be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #313, REVISION 0

ATTACHMENT C

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES, REVISION BAR FORMAT

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Plant Manager or his designee shall approve, prior to implementation, each proposed test, experiment or modifications to systems or equipment that affect nuclear safety.

5.1.2 The Shift Supervisor shall be responsible for the shift command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions responsible for activities affecting the safe handling and storage of nuclear fuel.

a. Lines of authority, responsibility, and communications shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These shall be documented in the FSAR;

b. The Decommissioning Director shall have overall responsibility for the safe handling and storage of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure the safe handling and storage of nuclear fuel. The Plant Manager shall be responsible to control those onsite activities necessary for the safe handling and storage of nuclear fuel; and

c. The individuals who train the Certified Fuel Handlers, carry out health physics or perform quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. Each duty shift shall be composed of at least one Shift Supervisor and one Non-certified Operator.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

b. Shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. At least one person qualified to stand watch in the control room (Non-certified Operator or Certified Fuel Handler) shall be present in the control room when nuclear fuel is stored in the spent fuel pools.

d. Oversight of fuel handling operations shall be provided by a Certified Fuel Handler.

e. The Shift Supervisor shall be a Certified Fuel Handler.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1, 1971 for comparable positions, except for the Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.2 A training and retraining program for the Certified Fuel Handler positions shall be maintained under the direction of the Plant Manager.

5.0 ADMINISTRATIVE CONTROLS

5.4 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.5 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.6 Procedures, Programs, and Manuals

5.6.1 Procedures

5.6.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. Quality assurance for effluent and environmental monitoring;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.6.2.

5.6.2 Programs and Manuals

The following programs shall be established, implemented, and maintained. Programs and Manuals may be titled as Reports.

5.6.2.1 Not Used

5.6.2.2 Not Used

5.6.2.3 Offsite Dose Calculation Manual (ODCM):

This Manual contains offsite dose calculation methodologies, the radioactive effluent controls program, and radiological environmental monitoring activities. The ODCM shall contain:

1. The methodologies and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents;

2. The methodologies and parameters used in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints;

3. The controls for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable in accordance with 10 CFR 50.36a. These include:

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.3 ODCM (continued)

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values of 10 CFR 20.1001 - 20.2401, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

    1. For noble gases: Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.3 ODCM (continued)

2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Licensee Initiated Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the on-site review function and the approval of the Plant Manager; and

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.3 ODCM (continued)

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date, (e.g., month/year) the change was implemented.

5.6.2.4 Not Used

5.6.2.5 Not Used

5.6.2.6 Not Used

5.6.2.7 Not Used

5.6.2.8 Not Used

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.9 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

b. Testing frequencies specified in the ASME OM Code and applicable Addenda;

c. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as two years or less in the Inservice Testing Program for performing inservice testing activities;

d. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.6.2.10 Not Used

5.6.2.11 Not Used

(continued)

5.6 Procedures, Programs and Manuals

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%.

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 Not Used

(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

5.6.2.16 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable); or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.16 SFDP (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.6.2.17 Technical Specifications (TS) Bases Control Program

Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to the review process specified in the Quality Assurance Plan.

Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

Proposed changes that meet the criteria of Specification 5.6.2.17.a or Specification 5.6.2.17.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.

5.6.2.18 Not Used

5.6.2.19 Not Used

5.6.2.20 Not Used

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.21 Control Complex Habitability Envelope Integrity Program

A Control Complex Habitability Envelope Integrity Program shall be established and implemented to ensure that CCHE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CCHE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a challenge from smoke. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CCHE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements.

1. The definition of the CCHE and the CCHE boundary.

2. Requirements for maintaining the CCHE boundary in its design condition including configuration control and preventive maintenance.

3. Requirements for (i) determining the unfiltered air in-leakage past the CCHE boundary into the CCHE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CCHE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

4. The Control Complex Habitability Envelope Integrity Program will be used to verify the integrity of the Control Complex boundary. Conditions that are identified to be adverse shall be trended and used as part of the 24 month assessment of the CCHE boundary.

5. The quantitative limits on unfiltered air in-leakage into the CCHE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air in-leakage measured by the testing described in paragraph 3. The unfiltered air in-leakage limit for radiological challenges is the in-leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air in-leakage limits for hazardous chemicals and smoke must ensure that exposure of CCHE occupants to these hazards will be within the assumptions in the licensing basis.

6. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CCHE habitability, determining CCHE unfiltered in-leakage as required by paragraph 3.

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

5.7.1.1 Reports required on an annual basis include:

a. Not Used

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM).

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

c. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted prior to May 1 of each year, and in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program, and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV B.1.

(continued)

5.6 Reporting Requirements

5.7.1.2 Not Used

5.7.2 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), alternative methods are used to control access to high radiation areas. Each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation (measured at 30 cm) is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance.

5.8.2 In addition to the requirements of Specification 5.8.1, areas with radiation levels ≥ 1000 mrem/hr at 30 cm shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor or health physics supervision. Doors shall remain locked except during periods of access by personnel.

Direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

(continued)

5.8 High Radiation Area

5.8.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that are not be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #313, REVISION 0

ATTACHMENT D

LIST OF REGULATORY COMMITMENTS

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Florida Power Corporation (FPC) in this document. Other statements in this correspondence are provided for information purposes and are not considered to be regulatory commitments. Please notify the Crystal River Unit 3 (CR-3) Licensing Supervisor of any questions regarding this document or any associated regulatory commitments.

Regulatory Commitment	Due date/event
CR-3 will vent and remove from service the Radioactive Waste System gas decay tanks.	August 30, 2013