Florida Power
A Progress Energy Company

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

Ref: 10 CFR 50.90

October 11, 2002
3F1002-03

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

Subject: Crystal River Unit 3 – License Amendment Request #272, Revision 0
Revision to Improved Technical Specifications 3.3.15 "Reactor Building (RB) Purge Isolation-High Radiation;" Bases 3.7.15 "Spent Fuel Assembly Storage;" 3.9.3 "Containment Penetrations;" and 3.9.6 "Refueling Canal Water Level"

Reference: Technical Specification Task Force (TSTF) Traveler 51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2

Dear Sir:

Pursuant to 10 CFR 50.90, Florida Power Corporation (FPC) hereby submits License Amendment Request (LAR) #272 which revises certain Crystal River Unit 3 (CR-3) Improved Technical Specifications to account for handling irradiated fuel within containment that has not occupied part of a critical reactor core within the previous 72 hours.

FPC requests NRC approval of LAR #272 by March 31, 2003, with implementation prior to entering MODE 6 for the Cycle 13 refueling outage. This implementation is similar to that approved for the Watts Bar Nuclear Plant, Unit 1 - Accession No. ML020100062.

This LAR implements the Nuclear Energy Institute (NEI) Technical Specification Task Force (TSTF) change traveler TSTF-51, Revision 2. TSTF-51, Revision 2, removes the technical specification applicability regarding operability of certain systems when handling fuel assemblies that have decayed a sufficient period of time such that dose consequences for the postulated fuel handling accident (FHA) remain below the 10 CFR 50.67 limits as determined by the CR-3 Alternate Source Term (AST). The systems removed from technical specification applicability include the containment penetrations and containment isolation. The CR-3 Improved Technical Specifications for the spent fuel assembly storage and the refueling canal water level are also revised by the TSTF-51, Revision 2.

Not all technical specifications revised by TSTF-51, Revision 2, are included in this LAR. Some TSTF-51, Revision 2, changes are not applicable to CR-3 since TSTF-51 was developed to modify NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants," Rev. 2, and the CR-3 ITS are based on Rev. 0 of that NUREG.

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CR-3 has determined that this request does not involve a significant hazards consideration pursuant to 10 CFR 50.92. In addition, there is no significant increase in the amounts of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an environmental assessment.

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

This letter establishes new regulatory commitments as contained in Attachment E.

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

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/rmb

Attachments:

A. Evaluation of License Amendment Request #272 - Introduction, Background, Description of Proposed Change, Reason for Request, Evaluation Of Request, References, and Precedents
B. Regulatory Analysis – No Significant Hazards Consideration Determination, Applicable Regulatory Requirements/Criteria, and Environmental Impact Evaluation
C. Proposed Revised Improved Technical Specifications and Bases Change Pages – Strikeout / Shadowed Format
D Proposed Revised Improved Technical Specifications and Bases Change Pages – Revision Bar Format
E. List of Regulatory Commitments

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

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President at the Crystal River Nuclear Plant for Progress Energy; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young
Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 11th day of October, 2002, by Dale E. Young

Lisa A Morris
Signature of Notary Public
State of Florida

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

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

Personally Known X -OR- Produced Identification _____

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT - 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #272
Implementation of TSTF-51, Rev. 2

Evaluation of License Amendment Request #272

Introduction, Background, Description of Proposed Change, Reason for Request, Evaluation of Request, References, and Precedents

EVALUATION OF LICENSE AMENDMENT REQUEST #272

Implementation of Technical Specification Task Force (TSTF) Item 51, "Revise Containment Requirements During Handling Irradiated Fuel And Core Alterations," Revision 2

Introduction

The purpose of this submittal is to request approval to implement TSTF-51.

Crystal River Unit 3 (CR-3) implemented the Improved Technical Specifications (ITS) based on NUREG-1430, B&W Standard Technical Specifications, Revision 0, in 1993. The Nuclear Energy Institute (NEI) Technical Specification Task Force has evaluated this change (TSTF-51, Revision 2) to the Standard Technical Specification NUREGs and has obtained NRC approval. This License Amendment Request (LAR) proposes to incorporate TSTF-51, Revision 2, into the CR-3 ITS.

Background

The CR-3 Alternate Source Term (AST) was approved by the NRC in License Amendment 199, dated September 17, 2001. The Fuel Handling Accident (FHA) at CR-3 was evaluated using the AST. FPC determined that the dose, as a result of the FHA after the reactor had been subcritical for 72 hours, was less than the dose limits of 10 CFR 50.67. The analysis assumed all radioactivity released into the containment was instantaneously released to the environment. No credit was taken for holdup in the containment or filtration of the release. Thus TSTF-51, Rev. 2, is applicable to CR-3 in that ITS for containment penetrations and containment isolation can be revised to not be applicable after the reactor has been subcritical for 72 hours.

Description of Proposed Change

Florida Power Corporation (FPC) proposes to revise the CR-3 ITS as supported by TSTF-51, Revision 2, as follows:

A. TSTF-51, Rev. 2, Changes that are not included in this LAR:

3.3.16 – Control Room Isolation – High Radiation (CR-3 ITS N/A) – This technical specification was deleted in License Amendment 199.

3.6.3 – Containment Isolation Valves (CR-3 ITS 3.6.3) – CR-3 does not open the Reactor Building Purge Valves in MODES 1, 2, 3, or 4.

3.7.10 – Control Room Emergency Ventilation System (CREVS) (CR-3 3.7.12) – Not applicable due to AST and License Amendment 199.

3.7.11 – Control Room Emergency Air Temperature Control System (CREATCS) (CR-3 ITS N/A) – Not in CR-3 ITS.

3.7.13 – Fuel Storage Pool Ventilation System (FSPVS) (CR-3 ITS N/A) – Not in CR-3 ITS.
3.8.2 – AC Sources – Shutdown (CR-3 ITS 3.8.2) – No CR-3 ITS Actions based on irradiated fuel.

3.8.5 – DC Sources – Shutdown (CR-3 ITS 3.8.5) – No CR-3 ITS Actions based on irradiated fuel.

3.8.8 – Inverters – Shutdown (CR-3 ITS 3.8.8) – No CR-3 ITS Actions based on irradiated fuel.

3.8.10 – Distribution Systems – Shutdown (CR-3 ITS 3.8.10) – No CR-3 ITS Actions based on irradiated fuel.

B. TSTF-51, Rev. 2, Changes that are included in this LAR (see also Attachment C for a Strikeout / Shadowed version of the proposed changes):

3.3.15 – RB Purge Isolation – High Radiation (CR-3 ITS 3.3.15)

ITS 3.3.15 Applicability was changed to read as the ITS 3.9.3 Applicability instead of referencing it and the word "recently" was added.

ITS 3.3.15 Required Action A.1 was changed to read as the ITS 3.9.3 Required Action instead of referencing them and the word "recently" was added.

ITS 3.3.15 Bases Background was changed to correct the terminology for the RB Purge Isolation – High Radiation Monitor and to allow the function to be bypassed as required by TSTF-51, Rev. 2.

ITS 3.3.15 Bases Applicable Safety Analyses was changed to include the time for irradiated fuel to be considered "recently" irradiated fuel and to delete the no longer needed allowance for open RB penetrations during movement of irradiated fuel.

ITS 3.3.15 Bases Applicability was changed to delete reference to LCO 3.9.3 and insert TSTF-51, Rev. 2, language.

ITS 3.3.15 Bases Actions was changed to match the changes to proposed ITS 3.3.15 Applicability and add TSTF-51, Rev. 2, wording.

3.7.16 – Spent Fuel Assembly Storage (CR-3 ITS 3.7.15)

ITS 3.7.15 Bases Actions was changed to be equivalent to TSTF-51, Rev. 2.

3.9.3 – Containment Penetrations (CR-3 ITS 3.9.3)

ITS 3.9.3 LCO was changed to specify actions on installed airlocks and to remove the allowance for open RB penetrations during movement of irradiated fuel.

ITS 3.9.3 Applicability was changed to implement TSTF-51, Rev. 2.

ITS 3.9.3 Actions were changed to implement TSTF-51, Rev. 2.

ITS 3.9.3 Bases Background was changed to implement TSTF-51, Rev. 2, and to remove the allowance for open RB penetrations during movement of irradiated fuel.

ITS 3.9.3 Bases Applicable Safety Analyses was changed to implement TSTF-51, Rev. 2, and to remove the allowance for open RB penetrations during movement of irradiated fuel.

ITS 3.9.3 Bases LCO was changed to implement TSTF-51, Rev. 2, and to remove the allowance for open RB penetrations during movement of irradiated fuel. Also, the wording in the last sentence of the first paragraph was revised to match ITS LCO 3.9.3 c.2 in only needing one valve to isolate the penetration and to match the title of ITS 3.3.15.

ITS 3.9.3 Bases Applicability was changed to implement TSTF-51, Rev. 2.

ITS 3.9.3 Bases Actions were changed to implement TSTF-51, Rev. 2.

ITS 3.9.3 Surveillance Requirements were changed to implement TSTF-51, Rev. 2, and to remove the allowance for open RB penetrations during movement of irradiated fuel.

3.9.6 – Refueling Canal Water Level (CR-3 ITS 3.9.6)

ITS 3.9.6 Applicability was changed to implement TSTF-51, Rev. 2.

ITS 3.9.6 Actions were changed to implement TSTF-51, Rev. 2.

ITS 3.9.6 Bases Background was changed to delete Core Alterations.

ITS 3.9.6 Bases Applicable Safety Analyses was changed to delete Core Alterations.

ITS 3.9.6 Bases Applicability was changed to delete Core Alterations.

ITS 3.9.6 Bases Actions were changed to delete Core Alterations.

Reason For Request

CR-3 is currently scheduled to begin Refuel 13 Outage in October 2003. With approval of this request, FPC can most efficiently schedule the movement of the replacement reactor vessel head and irradiated fuel movement. This change will also allow the uninterrupted movement of equipment into and out of the Reactor Building during this and future refueling outages.

Evaluation Of Request

This LAR implements TSTF-51, Revision 2, which has been approved by the NRC. The Alternate Source Term (AST) for CR-3 has been approved by the NRC. Using the CR-3 AST, the time for recently irradiated fuel to decay to a point where 10 CFR 50.67 dose limits are not exceeded due to an FHA is less than 72 hours.

CR-3 will install a contingency method to promptly close primary containment penetrations if an FHA occurs. Such prompt method need not completely block the penetration or be capable of resisting pressure. This includes a contingency method to cover the containment equipment hatch opening if it is not already closed. In addition, if containment penetrations to the environment are open during an FHA, the containment purge high radiation isolation will be bypassed and the containment purge exhaust or mini-purge started so that any radioactivity that might be released will be drawn in the proper direction and be monitored as it is being released.

References

The following documents were used in the development of this License Amendment Request (LAR):

1. Technical Specification Task Force Item 51, Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations, Revision 2.
2. NUREG-1430, Standard Technical Specifications Babcock and Wilcox Plants, Revision 2.
3. CR-3 Final Safety Analysis Report, Revision 27.
4. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, December 1991.
5. CR-3 License Amendment 199, Safety Evaluation Report, dated September 17, 2001.

Precedents

The NRC has approved similar submittals involving TSTF-51, Rev. 2:

Progress Energy	Brunswick	Accession No. ML020790479
Duke Power	Catawba	Accession No. ML021140431
FirstEnergy	Beaver Valley	Accession No. ML012330496
Florida Power & Light	St. Lucie	Accession No. ML022420403
Tennessee Valley Authority	Watts Bar	Accession No. ML020100062
Virginia Electric and Power	North Anna	Accession No. ML021200265

The NRC has received similar submittals involving TSTF-51, Rev. 2:

Progress Energy	Robinson 2	Accession No. ML022310271
Entergy	Indian Point 3	Accession No. ML021840136

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT - 3

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ATTACHMENT B

LICENSE AMENDMENT REQUEST #272
Implementation of TSTF-51, Rev. 2

Regulatory Analysis

No Significant Hazards Consideration Determination, Applicable Regulatory Requirements / Criteria, and Environmental Impact Evaluation

REGULATORY ANALYSIS

A. No Significant Hazards Consideration Determination

Crystal River Unit 3 (CR-3) proposes to revise Improved Technical Specifications (ITS) 3.3.15, 3.9.3, 3.9.6, and Bases 3.7.15.

Florida Power Corporation (FPC) has determined that this license amendment request does not involve a significant hazards consideration as defined in 10 CFR 50.92 based on the following:

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not increase the probability of a fuel handling accident in that the proposed change deals with the results of such an accident, not the cause of such an accident. The proposed change does not increase the consequences of an accident previously evaluated in that the CR-3 Alternate Source Term (AST) has been approved by the NRC, and this proposed change implements that NRC approval. The AST for the Fuel Handling Accident (FHA) takes no credit for containment isolation nor for a filtered release.

(2) Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the ITS do not affect nor create a different type of fuel handling accident. The fuel handling accident analyses assume that all of the iodine and noble gases that become airborne, escape, and reach the exclusion area boundary and low population zone with no credit taken for filtration, containment of the source term, or for decay or deposition in the containment. The proposed changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. The revised operations are consistent with the fuel handling accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does not involve a significant reduction in margin of safety.

The calculated doses to both the public and control room operators are well within the limits given in 10 CFR 50.67. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any ITS that is related to the establishment of or maintenance of a safety margin.

The systems that have been included in the proposed change will have administrative controls in place to assure that the systems are available and can be promptly returned to operation to further reduce dose consequences. These administrative controls will include a single normal or contingency method to promptly close the equipment hatch

opening. This prompt method need not completely block the hatch opening nor be capable of resisting pressure, but is to enable the ventilation systems to draw the release from the postulated FHA in the proper direction such that it can be monitored.

Therefore, operations of the facility in accordance with the proposed amendment would not involve a significant reduction in margin of safety.

Based on the above, FPC concludes that the proposed license amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

B. Applicable Regulatory Requirements / Criteria

This license amendment request conforms to Technical Specification Task Force (TSTF) Item 51, "Revise Containment Requirements During Handling Irradiated Fuel And Core Alterations," Revision 2, as applied to the Crystal River Unit 3 Improved Technical Specifications. Some TSTF-51, Rev. 2, changes are not applicable to CR-3 since TSTF-51 was developed to modify NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants," Rev. 2, and the CR-3 ITS are based on Rev. 0 of that NUREG. Other wording from TSTF-51 was incorporated into the CR-3 ITS to implement the intent of the TSTF-51.

In conformance with the TSTF-51, Rev. 2, FPC makes the following commitments:

1. Commitment:

   FPC will have procedures to require, in the case of a Fuel Handling Accident, a contingency method to promptly close primary containment penetrations that provide a path to the environment. Such prompt methods need not completely block the penetration or be capable of resisting pressure.

   Background:

   The purpose is to enable the Reactor Building Purge System Exhaust or the Reactor Building Mini-Purge to draw the radioactivity release from a postulated fuel handling accident in the proper direction such that it can be monitored by the Reactor Building Purge Radiation Monitor (RM-A1).

2. Commitment:

   FPC will have procedures in place which will, in the case of a Fuel Handling Accident, require the bypassing of the Containment Purge High Radiation Isolation and will require the initiation of the Containment purge exhaust or mini-purge exhaust so that any radioactivity that might be released by an FHA will be drawn in the proper direction and be monitored as it is being released.

Background:

Following shutdown, radioactivity in the Reactor Coolant System (RCS) decays fairly rapidly. The goal on maintaining either the Reactor Building Purge System Exhaust or the Reactor Building Mini-Purge, and the associated radiation monitor (RM-A1) availability, is to enable the ventilation system to draw the release for an FHA in the proper direction such that is can be monitored.

NUMARC 91-06 defines "availability" as the status of a system, structure, or component that is in service or can be placed in service in a functional or operable state by immediate manual or automatic actuation. "Functional" is defined as the ability of a system, structure, or component to perform its intended service with considerations that applicable Technical Specification requirements or licensing / design basis assumptions may not be maintained. "Operable" is defined as the ability of a system to perform its specified function with all applicable Technical Specification requirements satisfied.

C. 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 operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this license amendment request and has determined that this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the no significant hazards evaluation for this license amendment request (Attachment B).

2. The proposed change revises the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) as it relates to the approved Alternate Source Term (AST) for CR-3 and the Design Basis Fuel Handling Accident (FHA). Therefore, the proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site.

3. The proposed change involves purging the containment in the case of a FHA to remove any radioactivity that may be released. Therefore, the proposed license amendment will not result in a significant increase to the individual or cumulative occupational radiation exposure.

Therefore, pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT - 3

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ATTACHMENT C

LICENSE AMENDMENT REQUEST #272
Implementation of TSTF-51, Rev. 2

Proposed Revised Improved Technical Specifications and Bases Change Pages

Strikeout / Shadowed Format

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

3.3 INSTRUMENTATION

3.3.15 Reactor Building (RB) Purge Isolation-High Radiation

LCO 3.3.15 One channel of Reactor Building Purge Isolation-High Radiation shall be OPERABLE.

APPLICABILITY: ~~When containment purge or mini-purge valves are required to be OPERABLE by LCO 3.9.3, "Containment Penetrations."~~ During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 ~~Enter the applicable Conditions and Required Actions of LCO 3.9.3.~~ Suspend movement of recently irradiated fuel assemblies within containment	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.15.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.15.3	Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch or outage equipment hatch (OEH) installed and held in place by four bolts;

b. A minimum of one door in each installed air lock and the door in the OEH (if installed) closed~~, or capable of being closed by a designated individual readily available to close the open door~~; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent. ~~These penetrations may be open provided the total calculated flow rate out of the open penetration(s) is less than or equal to the equivalent flow rate through a 48 inch containment purge line penetration;~~ or

2. capable of being closed by an OPERABLE containment purge or mini-purge valve.

APPLICABILITY: ~~During CORE ALTERATIONS,~~
During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 156 ft Plant Datum.

APPLICABILITY: ~~During CORE ALTERATIONS,~~
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling canal water level not within limit.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~1 Suspend movement of irradiated fuel assemblies within containment.	Immediately
	AND	
	A.~~3~~2 Initiate action to restore refueling canal water level to within limit.	Immediately

B 3.3 INSTRUMENTATION

B 3.3.15 Reactor Building (RB) Purge Isolation-High Radiation

BASES

BACKGROUND

The RB Purge Isolation-High Radiation Function closes the RB purge and RB mini-purge valves to isolate the RB atmosphere from the environment and minimize releases of radioactivity in the event an accident occurs.

The radiation monitoring system (RM~~A-~~A1) measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The sensitive volume of the gas sampler is shielded with lead and monitored by a Geiger-Mueller detector. The air sample is taken from the center of the purge exhaust duct through a nozzle installed in the duct.

The monitor will alarm and initiate closure of the valves prior to exceeding the noble gas limits specified in the Offsite Dose Calculation Manual.

The closure of the purge and mini-purge valves ensures the RB remains as a barrier to fission product release. ~~There is no bypass for this function.~~

APPLICABLE SAFETY ANALYSES

FSAR Chapter 14 LOCA analysis assumes RB purge and mini purge lines are isolated within 60 seconds following initiation of the event. Since the early 1980's, this isolation time has only been practically applicable to the mini-purge valves since the large purge valves are required to be sealed closed during the MODES of plant operation (1, 2, 3, and 4) in which LOCAs are postulated to occur. Even

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

for mini-purge valves, design requirements on these valves require closure times on the order of 5 seconds. Thus, the purge isolation time of the current plant design is conservative to the original safety analysis.

The signal credited for initiating purge isolation in the original safety analysis is the RB Pressure - High ESAS signal and not RB Purge Isolation - High Radiation instrumentation. As such, design basis LOCA mitigation is not a basis for including this instrumentation.

RB purge isolation on high radiation is only required to maintain 10 CFR 20 limits during normal operations. However, this is not a basis for requiring a Technical Specification. Therefore, this Specification is not required in MODES 1, 2, 3 and 4.

Closure of the purge valves on high radiation is also not credited as part of the fuel handling accident (FHA) inside containment, which assumes fuel has decayed for 72 hours. The activity from the ruptured fuel assembly is assumed to be instantaneously released to the atmosphere in the form of a "puff" type release. Therefore, this specification is not required if moving fuel that has not been recently irradiated. ~~This instrumentation is retained during MODES 5 and 6 in order to allow for other RB penetrations that communicate with the RB atmosphere to be open during movement of irradiated fuel assemblies within containment. Refer to LCO 3.9.3, "Containment Penetrations" for further discussion of this allowance.~~

This specification is only required to minimize dose if moving fuel that has been recently irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours).

LCO

One channel of RB Purge Isolation-High Radiation instrumentation is required to be OPERABLE to ensure safety analysis assumptions regarding RB isolation are bounded. Operability of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

(continued)

BASES

APPLICABILITY

The RB Purge Isolation-High Radiation instrumentation shall be OPERABLE whenever movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 72 hours) within the RB is taking place. ~~required to support OPERABILITY of the purge and mini-purge valves, per LCO 3.9.3, "Containment Penetrations."~~ These ~~MODES and~~ specified conditions are indicative of those under which the potential for a fuel handling accident, ~~;~~ and thus radiation release, is the greatest. While in MODES 5 and 6, when ~~fuel~~ handling of recently irradiated fuel in the RB is not in progress, the isolation system does not need to be OPERABLE because the potential for a significant radioactive release is minimal and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

ACTIONS

A.1

Condition A applies to failure of the high radiation purge isolation function during movement of recently irradiated fuel assemblies within containment. ~~when the purge and mini-purge valves are required to be OPERABLE in accordance with LCO 3.9.3, "Containment Penetrations."~~

~~With the channel inoperable during this time, the applicable Conditions and Required Actions of LCO 3.9.3 are required to be entered immediately.~~ The immediate Completion Time is consistent with the loss of RB isolation capability under conditions in which the fuel handling accidents involving handling recently irradiated fuel are possible and the high radiation function is required to provide automatic action to terminate the release.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR is the performance of the CHANNEL CHECK for the RB purge isolation-high radiation instrumentation once every12 hours. The CHANNEL CHECK is a comparison of the parameter indicated on the radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of

(continued)

BASES

LCO (continued)

Fuel with burnup-enrichment combinations in the area above the upper curve has no restrictions on where it can be stored. Fuel with burnup-enrichment combinations in the area between the lower and upper curves must be stored in the peripheral cells of the pool. The peripheral cells are those that are adjacent to the walls of the spent fuel pool. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in Pool B, but must be stored in Pool A.

The LCO allows compensatory loading techniques, specified in the FSAR and applicable fuel handling procedures, as an alternative to storing fuel assemblies in accordance with Figures 3.7.15-1 and 3.7.15-2. This is acceptable since these loading patterns assure the same degree of subcriticality within the pool.

APPLICABILITY

In general, limiting fuel enrichment of stored fuel prevents inadvertent criticality in the storage pools. Inadvertent criticality is dependent on whether fuel is stored in the pools and is completely independent of plant MODE.

Therefore, this LCO is applicable whenever any fuel assembly is stored in high density fuel storage locations.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating LCO 3.0.3 does not apply. Since the design basis accident of concern in this Specification is an inadvertent criticality, and since the possibility or consequences of this event are independent of plant MODE, there is no reason to shutdown the plant if the LCO or Required Actions cannot be met.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1 or Figure 3.7.15-2, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance. The Immediate Completion Time underscores the necessity of restoring spent fuel pool ~~irradiated~~ fuel loading to within the initial assumptions of the criticality analysis.

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

An accident which occurs during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment will have any released radioactivity limited from escaping to the environment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, the requirement to isolate the containment from the outside atmosphere is less stringent than those established for MODES 1 through 4. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths for radioactivity are closed or capable of being closed by an OPERABLE containment purge or mini-purge valve.

The containment equipment hatch or outage equipment hatch (OEH) provides a means for moving large equipment and components into and out of containment. During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, the equipment hatch or OEH must be held in place by at least four bolts. The required number of bolts is based on dead weight and is acceptable due to the low likelihood of a pressurization event. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door in the OEH (if installed) must always remain closed ~~or be capable of being closed~~.

The containment air locks provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. However, during periods of unit shutdown when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an installed air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed ~~or be capable of being closed~~.

(continued)

BASES

BACKGROUND (continued)

~~If the door in the OEH (if installed) or both doors in the containment air locks are open when containment closure is required, a designated individual must be readily available to close the door in the OEH and at least one door in each air lock. Operations personnel directly involved in refueling operations shall be aware of the identity of the designated individual(s). The designated individual(s) shall remain within sufficient proximity to the open doors to assist in evacuation of personnel inside containment and to close the open door(s) as soon as evacuation is completed.~~

The requirements on containment penetration closure ensure that a release of fission product radioactivity to the environment from the containment will be limited. The closure restrictions are sufficient to limit fission product radioactivity release from containment due to a fuel handling accident involving handling recently irradiated fuel during refueling.

In MODE 6, it is necessary to periodically recirculate/exchange RB atmosphere in order to minimize radiation uptake during the conduct of refueling operations. The 48 inch purge valves are normally used for this purpose, but the mini-purge valves may be relied upon as well. Both valve types are automatically isolated on a unit vent-high radiation signal (from RM-A1). So long as one valve in the flow path is OPERABLE, these lines may remain unisolated during the subject plant conditions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by a minimum of one isolation device. Isolation may be achieved by an automatic or manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material (e.g., temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements involving handling recently irradiated fuel.

~~These penetrations may be open provided the total calculated flow rate out the open penetrations is less than or equal to the equivalent flow rate through a 48 inch containment purge line penetration. This allowance is consistent with the CR-3 fuel handling accident inside the reactor building. The licensing basis analysis assumed a puff release of radionuclides from the RB following the FHA event. No credit was taken for the RB purge filters. Limiting the flow rate out the open penetrations to a flow rate less than or equal to the flow rate through the RB purge system is reasonable and conservative, given the plant licensing basis. Offsite doses from this analysis are within 10 CFR 50.67 limits. With the containment purge valves OPERABLE, no leakage value has to be assigned to these penetrations, and the entire~~

(continued)

BASES

BACKGROUND (continued)

~~50,000 cfm can be allocated to other penetrations providing direct access. With the containment purge valves inoperable, these valves are allowed to be open during the Applicability of this Specification, however; no additional penetrations are allowed to be un-isolated during this time.~~

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment ~~with irradiated fuel in containment~~, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Canal Water Level," in conjunction with the administrative limit on minimum decay time of 72 hours prior to ~~the movement of irradiated fuel in the vessel, and this LCO~~ irradiated fuel movement ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 50.67 even without containment closure.

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere, including the equipment hatch or the Outage Equipment Hatch, to be closed except for penetrations containing an OPERABLE purge or mini-purge valve. ~~For the containment air locks and the OEH (if installed), both doors in the air locks and the door in the OEH may be open only under administrative controls.~~ For the containment purge and mini-purge valves to be considered OPERABLE, at least one ~~these~~ valves in each ~~(~~penetration~~s)~~ must be automatically isolable on an RB Purge~~unit vent~~-high radiation isolation signal.

The definition of "direct access from the containment atmosphere to the outside atmosphere" is any path that would allow for transport of containment atmosphere to any atmosphere located outside the containment structure. This includes the Auxiliary Building. As a general rule, closed or pressurized systems do not constitute a direct path

(continued)

BASES

LCO (continued)

between the RB and outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.

~~These penetrations may be open provided the total calculated flow rate out the open penetrations is less than or equal to the equivalent flow rate through a 48 inch containment purge line penetration.~~

APPLICABILITY

The containment penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment because this is ~~the period of highest risk~~ when there is a potential for ~~a~~ the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, w~~W~~hen ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment ~~are~~is not being conducted; the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving fuel that has not been recently irradiated (i.e., fuel that has not occupied part of a critical reactor core within the previous 72 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 ~~and A.2~~

With the containment equipment hatch, OEH, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including ~~failure to implement required administrative controls for open OEH and air lock doors and~~ the containment purge or mini-purge valve penetrations not capable of automatic isolation when the penetrations are unisolated, the plant must be placed in a condition in which the isolation function is not

(continued)

BASES

ACTIONS

A.1~~ and A.2~~ (continued)

needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position~~, and that administrative controls required for open OEH and air lock doors are being implemented~~.

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

As such, this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity within containment will not result in a release of significant fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and mini-purge valve actuates to its isolation position on an actual or simulated high radiation signal. The 24 month Frequency is consistent with other similar instrumentation and valve testing requirements. The Surveillance ensures that the valves are capable of closing after a postulated fuel handling accident involving handling recently irradiated fuel to limit a release of fission product radioactivity from the containment. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.

REFERENCES

1. FSAR, Section 14.2.2.3.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment ~~or performance of CORE ALTERATIONS~~ requires a minimum refueling canal water level of 156 ft plant datum. This maintains sufficient water level above the fuel contained in the vessel and the bottom of the fuel transfer canal, and the spent fuel pool to ensure iodine fission product activity is retained in the water to a level consistent with the dose analysis of a fuel handling accident (Ref. 4). Sufficient iodine activity would be retained to limit offsite doses from the accident to well within 10 CFR 50.67 limits (Ref. 3).
APPLICABLE SAFETY ANALYSES	During ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies, the water level in the refueling canal is an assumed initial condition in the analysis of the fuel handling accident in containment. This relates to the assumption that 99% of the total iodine released from the fuel is retained by the refueling canal water. There are postulated drop scenarios where there is < 23 ft above the top of the fuel bundle and the surface. In particular, this is the case for the period of time during which the assembly travels between the cavity and the deep end of the refueling canal. During this time, there is potentially 21 feet of water between the reactor vessel flange (135 ft plant datum) and the surface of the pool. The iodine retention factors used in the dose assessment are still conservative at water levels of 21 feet above the damaged fuel (Ref. 4). The 156 ft value was chosen to be consistent with the level specified for LCO 3.7.13, "Fuel Storage Pool Water Level" and plant configuration.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel handling accident analysis inside containment is described in Reference 4. With a minimum water level of 23 ft above the stored fuel, and the administrative limit on minimum decay time of 72 hours prior to movement of irradiated fuel in the vessel, analyses demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water such that offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

A minimum refueling canal water level of 156 ft plant datum is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits. This minimum level also ensures an adequate operational window between the surface of the pool and the transfer winch for the RB fuel handling equipment.

APPLICABILITY

This Specification is applicable ~~during CORE ALTERATIONS and~~ when moving irradiated fuel assemblies within the containment. The LCO minimizes the potential of a fuel handling accident in containment which results in offsite doses greater than those calculated by the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Water level requirements for fuel handling accidents postulated to occur in the spent fuel pool are addressed by LCO 3.7.13, "Fuel Storage Pool Water Level."

ACTIONS

A.1~~, A.2 and A.3~~

With a refueling canal water level of < 156 ft plant datum, all ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies shall be suspended immediately to preclude a fuel handling accident from occurring. The suspension of ~~CORE ALTERATIONS and~~ fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES

ACTIONS (continued)

~~A.1, A.2 and A.3 (continued)~~

In addition to immediately suspending ~~CORE ALTERATIONS or~~ movement of irradiated fuel, actions to restore refueling canal water level must be initiated immediately. The immediate Completion Time is based on engineering judgment. When increasing refueling canal water level, the boron concentration of the make-up and the effect of this concentration on the minimum specified in the COLR (Ref. LCO 3.9.1) must be considered.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum refueling canal water level of 156 ft plant datum ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are assumed to result from a postulated fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Deleted.

2. FSAR Section 14.2.2.3.

3. 10 CFR 50.67.

4. FPC Calculation N-00-0001.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT - 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT D

LICENSE AMENDMENT REQUEST #272
Implementation of TSTF-51, Rev. 2

Proposed Revised Improved Technical Specifications and Bases Change Pages

Revision Bar Format

3.3 INSTRUMENTATION

3.3.15 Reactor Building (RB) Purge Isolation-High Radiation

LCO 3.3.15 One channel of Reactor Building Purge Isolation-High Radiation shall be OPERABLE.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.15.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.15.3	Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch or outage equipment hatch (OEH) installed and held in place by four bolts;

b. A minimum of one door in each installed air lock and the door in the OEH (if installed) closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent; or

2. capable of being closed by an OPERABLE containment purge or mini-purge valve.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 156 ft Plant Datum.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling canal water level not within limit.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately
	AND	
	A.2 Initiate action to restore refueling canal water level to within limit.	Immediately

B 3.3 INSTRUMENTATION

B 3.3.15 Reactor Building (RB) Purge Isolation-High Radiation

BASES

BACKGROUND

The RB Purge Isolation-High Radiation Function closes the RB purge and RB mini-purge valves to isolate the RB atmosphere from the environment and minimize releases of radioactivity in the event an accident occurs.

The radiation monitoring system (RMA-1) measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The sensitive volume of the gas sampler is shielded with lead and monitored by a Geiger-Mueller detector. The air sample is taken from the center of the purge exhaust duct through a nozzle installed in the duct.

The monitor will alarm and initiate closure of the valves prior to exceeding the noble gas limits specified in the Offsite Dose Calculation Manual.

The closure of the purge and mini-purge valves ensures the RB remains as a barrier to fission product release.

APPLICABLE SAFETY ANALYSES

FSAR Chapter 14 LOCA analysis assumes RB purge and mini-purge lines are isolated within 60 seconds following initiation of the event. Since the early 1980's, this isolation time has only been practically applicable to the mini-purge valves since the large purge valves are required to be sealed closed during the MODES of plant operation (1, 2, 3, and 4) in which LOCAs are postulated to occur. Even

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

for mini-purge valves, design requirements on these valves require closure times on the order of 5 seconds. Thus, the purge isolation time of the current plant design is conservative to the original safety analysis.

The signal credited for initiating purge isolation in the original safety analysis is the RB Pressure - High ESAS signal and not RB Purge Isolation - High Radiation instrumentation. As such, design basis LOCA mitigation is not a basis for including this instrumentation.

RB purge isolation on high radiation is only required to maintain 10 CFR 20 limits during normal operations. However, this is not a basis for requiring a Technical Specification. Therefore, this Specification is not required in MODES 1, 2, 3 and 4.

Closure of the purge valves on high radiation is also not credited as part of the fuel handling accident (FHA) inside containment, which assumes fuel has decayed for 72 hours. The activity from the ruptured fuel assembly is assumed to be instantaneously released to the atmosphere in the form of a "puff" type release. Therefore, this specification is not required if moving fuel that has not been recently irradiated.

This specification is only required to minimize dose if moving fuel that has been recently irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours).

LCO

One channel of RB Purge Isolation-High Radiation instrumentation is required to be OPERABLE to ensure safety analysis assumptions regarding RB isolation are bounded. Operability of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

(continued)

BASES

APPLICABILITY

The RB Purge Isolation-High Radiation instrumentation shall be OPERABLE whenever movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours) within the RB is taking place. These specified conditions are indicative of those under which the potential for a fuel handling accident, and thus radiation release, is the greatest. While in MODES 5 and 6, when handling of recently irradiated fuel in the RB is not in progress, the isolation system does not need to be OPERABLE because the potential for a significant radioactive release is minimal and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

ACTIONS

A.1

Condition A applies to failure of the high radiation purge isolation function during movement of recently irradiated fuel assemblies within containment.

The immediate Completion Time is consistent with the loss of RB isolation capability under conditions in which the fuel handling accidents involving handling recently irradiated fuel are possible and the high radiation function is required to provide automatic action to terminate the release.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR is the performance of the CHANNEL CHECK for the RB purge isolation-high radiation instrumentation once every 12 hours. The CHANNEL CHECK is a comparison of the parameter indicated on the radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of

(continued)

BASES

LCO (continued)

Fuel with burnup-enrichment combinations in the area above the upper curve has no restrictions on where it can be stored. Fuel with burnup-enrichment combinations in the area between the lower and upper curves must be stored in the peripheral cells of the pool. The peripheral cells are those that are adjacent to the walls of the spent fuel pool. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in Pool B, but must be stored in Pool A.

The LCO allows compensatory loading techniques, specified in the FSAR and applicable fuel handling procedures, as an alternative to storing fuel assemblies in accordance with Figures 3.7.15-1 and 3.7.15-2. This is acceptable since these loading patterns assure the same degree of subcriticality within the pool.

APPLICABILITY

In general, limiting fuel enrichment of stored fuel prevents inadvertent criticality in the storage pools. Inadvertent criticality is dependent on whether fuel is stored in the pools and is completely independent of plant MODE.

Therefore, this LCO is applicable whenever any fuel assembly is stored in high density fuel storage locations.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating LCO 3.0.3 does not apply. Since the design basis accident of concern in this Specification is an inadvertent criticality, and since the possibility or consequences of this event are independent of plant MODE, there is no reason to shutdown the plant if the LCO or Required Actions cannot be met.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1 or Figure 3.7.15-2, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance. The Immediate Completion Time underscores the necessity of restoring spent fuel pool fuel loading to within the initial assumptions of the criticality analysis.

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

An accident which occurs during movement of recently irradiated fuel assemblies within containment will have any released radioactivity limited from escaping to the environment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, the requirement to isolate the containment from the outside atmosphere is less stringent than those established for MODES 1 through 4. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths for radioactivity are closed or capable of being closed by an OPERABLE containment purge or mini-purge valve.

The containment equipment hatch or outage equipment hatch (OEH) provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch or OEH must be held in place by at least four bolts. The required number of bolts is based on dead weight and is acceptable due to the low likelihood of a pressurization event. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door in the OEH (if installed) must always remain closed.

The containment air locks provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. However, during periods of unit shutdown when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an installed air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

(continued)

BASES

BACKGROUND (continued)

The requirements on containment penetration closure ensure that a release of fission product radioactivity to the environment from the containment will be limited. The closure restrictions are sufficient to limit fission product radioactivity release from containment due to a fuel handling accident involving handling recently irradiated fuel during refueling.

In MODE 6, it is necessary to periodically recirculate/exchange RB atmosphere in order to minimize radiation uptake during the conduct of refueling operations. The 48 inch purge valves are normally used for this purpose, but the mini-purge valves may be relied upon as well. Both valve types are automatically isolated on a unit vent-high radiation signal (from RM-A1). So long as one valve in the flow path is OPERABLE, these lines may remain unisolated during the subject plant conditions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by a minimum of one isolation device. Isolation may be achieved by an automatic or manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material (e.g., temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements involving recently irradiated fuel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

During movement of recently irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Canal Water Level," in conjunction with the administrative limit on minimum decay time of 72 hours prior to irradiated fuel movement ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 50.67 even without containment closure.

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere, including the equipment hatch or the Outage Equipment Hatch, to be closed except for penetrations containing an OPERABLE purge or mini-purge valve. For the containment purge and mini-purge valves to be considered OPERABLE, at least one valve in each penetration must be automatically isolable on an RB Purge-high radiation isolation signal.

The definition of "direct access from the containment atmosphere to the outside atmosphere" is any path that would allow for transport of containment atmosphere to any atmosphere located outside the containment structure. This includes the Auxiliary Building. As a general rule, closed or pressurized systems do not constitute a direct path

(continued)

BASES

LCO (continued)

between the RB and outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving fuel that has not been recently irradiated (i.e., fuel that has not occupied part of a critical reactor core within the previous 72 hours) will result in doses that are will within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

With the containment equipment hatch, OEH, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the containment purge or mini-purge valve penetrations not capable of automatic isolation when the penetrations are unisolated, the plant must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

As such, this surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity with containment will not result in a release of significant fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and mini-purge valve actuates to its isolation position on an actual or simulated high radiation signal. The 24 month Frequency is consistent with other similar instrumentation and valve testing requirements. The Surveillance ensures that the valves are capable of closing after a postulated fuel handling accident involving handling recently irradiated fuel to limit a release of fission product radioactivity from the containment. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.

REFERENCES

1. FSAR, Section 14.2.2.3.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum refueling canal water level of 156 ft plant datum. This maintains sufficient water level above the fuel contained in the vessel and the bottom of the fuel transfer canal, and the spent fuel pool to ensure iodine fission product activity is retained in the water to a level consistent with the dose analysis of a fuel handling accident (Ref. 4). Sufficient iodine activity would be retained to limit offsite doses from the accident to well within 10 CFR 50.67 limits (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal is an assumed initial condition in the analysis of the fuel handling accident in containment. This relates to the assumption that 99% of the total iodine released from the fuel is retained by the refueling canal water. There are postulated drop scenarios where there is < 23 ft above the top of the fuel bundle and the surface. In particular, this is the case for the period of time during which the assembly travels between the cavity and the deep end of the refueling canal. During this time, there is potentially 21 feet of water between the reactor vessel flange (135 ft plant datum) and the surface of the pool. The iodine retention factors used in the dose assessment are still conservative at water levels of 21 feet above the damaged fuel (Ref. 4). The 156 ft value was chosen to be consistent with the level specified for LCO 3.7.13, "Fuel Storage Pool Water Level" and plant configuration.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel handling accident analysis inside containment is described in Reference 4. With a minimum water level of 23 ft above the stored fuel, and the administrative limit on minimum decay time of 72 hours prior to movement of irradiated fuel in the vessel, analyses demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water such that offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

A minimum refueling canal water level of 156 ft plant datum is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits. This minimum level also ensures an adequate operational window between the surface of the pool and the transfer winch for the RB fuel handling equipment.

APPLICABILITY

This Specification is applicable when moving irradiated fuel assemblies within the containment. The LCO minimizes the potential of a fuel handling accident in containment which results in offsite doses greater than those calculated by the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Water level requirements for fuel handling accidents postulated to occur in the spent fuel pool are addressed by LCO 3.7.13, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a refueling canal water level of < 156 ft plant datum, all movement of irradiated fuel assemblies shall be suspended immediately to preclude a fuel handling accident from occurring. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES

ACTIONS

A.2

In addition to immediately suspending movement of irradiated fuel, actions to restore refueling canal water level must be initiated immediately. The immediate Completion Time is based on engineering judgment. When increasing refueling canal water level the boron concentration of the make-up and the effect of this concentration on the minimum specified in the COLR (Ref. LCO 3.9.1) must be considered.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum refueling canal water level of 156 ft plant datum ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are assumed to result from a postulated fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Deleted.

2. FSAR Section 14.2.2.3.

3. 10 CFR 50.67.

4. FPC Calculation N-00-0001.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT - 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT E

LICENSE AMENDMENT REQUEST #272
Implementation of TSTF-51, Rev. 2

List of Regulatory Commitments

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Florida Power Corporation (FPC) in this document. Any other actions discussed in the submittal represent intended or planned actions by FPC. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

Commitment	Due Date
FPC will have procedures to require, in the case of a Fueling Handling Accident, a contingency method to promptly close primary containment penetrations that provide a path to the environment. Such prompt methods need not completely block the penetrations or be capable of resisting pressure.	Prior to entering MODE 6 for the Cycle 13 refueling outage.
FPC will have procedures in place which will, in the case of a Fuel Handling Accident, require the bypassing of the Containment Purge High Radiation Isolation and will require the initiation of the Containment purge exhaust or mini-purge exhaust so that any radioactivity that might be released by an FHA will be drawn in the proper direction and be monitored as it is being released.	Prior to entering MODE 6 for the Cycle 13 refueling outage.