Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

December 10, 2009

L-2009-282
10 CFR 50.4

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

RE: St. Lucie Unit 2
Docket No. 50-389
Technical Specification Bases Control Program
Periodic Report of Bases Changes TS 6.8.4.j.4

Pursuant to Technical Specification (TS) 6.8.4.j.4, Florida Power & Light Company (FPL) is submitting the periodic report of changes made to the St. Lucie Unit 2 TS Bases without prior NRC approval. The requirement for the periodic report was added by St. Lucie Unit 2 License Amendment 117 on July 12, 2001 and is required on a frequency consistent with 10 CFR 50.71(e) for UFSAR updates. FPL submits the 10 CFR 50.71(e) reports within six months of the completion of each refueling outage. This periodic report covers the period from April 4, 2008 to the startup from the spring 2009 Unit 2 refueling outage (SL2-18).

FPL is submitting the current revision of ADM-25.04, St. Lucie Unit 2 Technical Specification Bases Attachments 1 through 13. Each attachment summarizes the revisions on the attachment cover page.

Please contact us if there are any questions regarding this submittal.

Sincerely,

E S Katzman

Eric S. Katzman
Licensing Manager
St. Lucie Plant

ESK/tlt

Attachments

A001
NRR

FPL	ST. LUCIE PLANT ADMINISTRATIVE PROCEDURE NON-SAFETY RELATED **INFORMATION USE**	Procedure No. **ADM-25.04** Current Revision No. **29** Effective Date **11/12/09**

Title:

ST. LUCIE PLANT TECHNICAL SPECIFICATIONS BASES CONTROL PROGRAM AND TECHNICAL SPECIFICATIONS BASES

Responsible Department: **LICENSING**

REVISION SUMMARY:

Revision 29 - Incorporated PCR 08-0501 for CR 2007-32178 to revise TS Bases to implement AST License Amendment. Procedure changes were reviewed in ORG 07-041 on 5/29/07 as part of the license amendment submittal. (Author: Ken Frehafer)

Revision 28 - Incorporated PCR 09-2643 to update EDG fuel oil testing ASTM standards. (Author: Ken Frehafer)

Revision 27 - Incorporated PCR 08-6765 for CR 2008-32178 to change TS Bases to implement Unit 2 AST LAR. (Author: Ken Frehafer)

AND

Incorporated PCR 08-6764 for CR 2008-23612 to revise Unit 1 and 2 TS Bases to implement CR Habitability LAR. (Author: Ken Frehafer)

Revision 26 - Incorporated PCR 08-6762 for CR 2008-17239 to implement License Amendment 207 and 155. Procedure changes to implement EDG Fuel Oil Test Program LAR were reviewed in ORG 08-034 on 6/26/08 as part of the license amendment submittal. (Author: K.W. Frehafer)

Revision 25 - Incorporated PCR 09-1217 for CR 2009-4976 to incorporate TSTF-434 - Overlap Testing - in Bases for SR 4.0.1. (Author: Ken Frehafer)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R. G. West Plant General Manager	08/30/01
Revision	**FRG Review Date**	**Approved By**	**Approval Date**
29		N/A Plant General Manager	
		Eric Katzman Authorized Approver	09/23/09
		N/A Authorized Approver (Minor Correction)	

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DOCT	PROCEDURE
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1.0 PURPOSE

1.1 This procedure provides instructions for the preparation, review, approval, distribution, revision, and cancellation changes to the BASES of the Technical Specifications as required by St. Lucie Unit 1 and Unit 2 Technical Specification 6.8.4.j.

1.2 BASES changes are not a substitute for a License Amendment. The discussion provided in the BASES cannot change the meaning or intent of the Technical Specifications. The BASES can only provide guidance in what is necessary to meet the intent of the Technical Specifications.

1.3 This procedure implements the Technical Specification requirements of St. Lucie Unit 1 and Unit 2 Technical Specification 6.8.4.j, "BASES Control Program," that states:

1. This program provides a means for processing changes to the Bases of these Technical Specifications.

    A. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

    B. Changes may be made to Bases without prior NRC approval provided the changes do not require either of the following:

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

    2. A change to the updated UFSAR or Bases that requires NRC approval pursuant to10 CFR 50.59.

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

    D. Proposed changes that meet the criteria of Technical Specification 6.8.4.j.2.a or 6.8.4.j.2.b. (step 1.3.1.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(e).

2.0 REFERENCES

NOTE

One or more of the following symbols may be used in this procedure:

§ Indicates a Regulatory commitment made by Technical Specifications, Condition of License, Audit, LER, Bulletin, Operating Experience, License Renewal, etc. and shall NOT be revised without the required Focus review and appropriate approval.

¶ Indicates a management directive, vendor recommendation, plant practice or other non-regulatory commitment that should NOT be revised without consultation with the plant staff.

Ψ Indicates a step that requires a sign off on an attachment.

2.1 Quality Instructions / Plant Procedures

- QI-5-PSL-1, Preparation, Revision, Review / Approval of Procedures
- ENG-QI 2.0, Engineering Evaluation
- ENG-QI 2.1, 10 CFR 50.59 Screening / Evaluation
- ADM-17.10, Processing Engineering Evaluations
- NAP-409, Processing of Proposed or Approved License Amendments

2.2 Regulations and Regulatory Guidelines

- NUREG-1432, Rev 1, Combustion Engineering Standard Technical Specifications
- 10 CFR 50.59, Changes, Tests and Experiments
- NSAC-125, Guidelines for 10 CFR 50.59 Safety Evaluations
- 10 CFR 50.71, Maintenance of records, making of reports
- 10 CFR 50.36, Technical specifications
- St. Lucie Unit 1 Operating License Amendment
- St. Lucie Unit 2 Operating License Amendment
- Technical Specification 6.8.4.j

2.3 Miscellaneous Documents (i.e., PC/M, Correspondence)

3.0 RESPONSIBILITIES

3.1 The Plant General Manager is responsible for approval of all Technical Specification BASES changes.

3.2 The Facility Review Group (FRG) is responsible for review and recommending approval or disapproval of all Technical Specification BASES changes.

3.3 The Operations Manager is responsible for reviewing the Technical Specification BASES changes for plant operational impact.

3.4 The Licensing Manager is responsible for:

- The overall implementation of the Technical Specification BASES Control Program
- Submission to the NRC of changes to the Technical Specification BASES on the same schedule as the periodic update to the UFSAR as required by 10 CFR 50.71(e).

3.5 The individual responsible for proposed changes to the Technical Specification BASES shall process the proposed change in accordance with QI-5-PSL-1.

4.0 DEFINITIONS

4.1 **50.59 Evaluation** -The record required by 10 CFR 50.59, paragraph (b) that provides the basis for determination that the change, test or experiment does not require prior NRC approval. For those activities that do not require prior NRC approval, the 50.59 evaluation serves to document and justify the change does not require prior NRC approval. The document should record the scope of the evaluation and the logic for the determination that NRC prior approval is not required.

4.2 **Technical Specification BASES** - A set of documentation providing elaboration and interpretation of the Technical Specifications and their application to physical systems in the plant.

5.0 RECORDS REQUIRED

5.1 Completed documents, or Similar Forms, required by QI-5-PSL-1 shall be maintained in the plant files in accordance with QI-17-PSL-1, Quality Assurance Records.

6.0 INSTRUCTIONS

6.1 Changes to the Technical Specification BASES shall be proposed as a revision to this procedure in accordance with the plant's procedure change process specified in QI-5-PSL-1.

6.2 Proposed changes to the Technical Specification BASES should take into consideration the BASES for the similar specification (if one exists) in NUREG-1432, Rev 1, Combustion Engineering Standard Technical Specifications and BASES thereto as well as the St. Lucie Unit 1 or St. Lucie Unit 2 Updated Final Safety Analysis Report, Design Basis Documents and applicable NRC Correspondence, as applicable.

6.3 If the answers to all the 10 CFR 50.59 Evaluation Checklist Safety Review questions are No, the proposed BASES and procedure change may proceed.

6.4 If any of the 10 CFR 50.59 Evaluation Checklist Safety Review questions is checked Yes, a safety evaluation is required, and shall be attached to the BASES change prior to submittal for review by the FRG and approval by the Plant General Manager.

6.5 If the BASES change is determined to NOT be able to be made pursuant to 10 CFR 50.59 or the BASES change also requires a change to the Technical Specifications, the change shall be submitted to the NRC, in accordance with 10 CFR 50.90 and NAP-409, Processing of Proposed or Approved License Amendments, for approval prior to implementation.

6.6 Each section of the Technical Specification BASES (e.g., the BASES associated with Technical Specification 3/4.5, or 3/4.8) shall have the same revision number, regardless of the extent of the revision.

6.7 The current revision of each specific Technical Specification BASES attachment shall be listed in this procedure. Revisions to the BASES will be performed by revising this procedure and the applicable section of the BASES. BASES sections that are not revised will remain unchanged in content and revision number.

6.8 The current revision number for each page of the BASES is identified by the revision number on each page and shall be the same as the effective revision for that BASES section listed in Appendix A and Appendix B to this procedure.

6.9 Appendix A and Appendix B shall list the effective revision of each BASES section.

6.10 Each BASES page shall be marked "UNIT 1" or "UNIT 2" and shall be numbered "page x of y."

6.11 Upon FRG and Plant General Manager approval of revisions to ADM-25.04, the revised procedure and only the revised attachment(s) of ADM-25.04 shall be distributed.

6.12 Revised changes to the Technical Specification BASES implemented in ADM-25.04 shall be distributed in accordance with QI-6-PSL-1, Document Control.

END OF SECTION 6.0

APPENDIX A
ST. LUCIE UNIT 1 TECHNICAL SPECIFICATION BASES
(Page 1 of 1)

Attachment	Title	Revision
1	BASES for Section 2.0 – SAFETY LIMITS AND LIMITING SAFETY SETTINGS	1
2	BASES for Sections 3.0 and 4.0 – LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS	2
3	BASES for Sections 3/4.1 – REACTIVITY CONTROL SYSTEMS	2
4	BASES for Sections 3/4.2 – POWER DISTRIBUTION LIMITS	0
5	BASES for Sections 3/4.3 – INSTRUMENTATION	1
6	BASES for Sections 3/4.4 – REACTOR COOLANT SYSTEM	4
7	BASES for Sections 3/4.5 – EMERGENCY CORE COOLING SYSTEMS (ECCS)	2
8	BASES for Sections 3/4.6 – CONTAINMENT SYSTEMS	6
9	BASES for Sections 3/4.7 – PLANT SYSTEMS	3
10	BASES for Sections 3/4.8 – ELECTRICAL POWER SYSTEMS	3
11	BASES for Sections 3/4.9 – REFUELING OPERATIONS	5
12	BASES for Sections 3/4.10 – SPECIAL TEST EXCEPTIONS	0
13	BASES for Sections 3/4.11 – RADIOACTIVE EFFLUENTS	0

/R29

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END OF APPENDIX

APPENDIX B
ST. LUCIE UNIT 2 TECHNICAL SPECIFICATION BASES
(Page 1 of 1)

Attachment	Title	Revision
1	BASES for Section 2.0 – SAFETY LIMITS AND LIMITING SAFETY SETTINGS	4
2	BASES for Sections 3.0 and 4.0 – LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS	2
3	BASES for Sections 3/4.1 – REACTIVITY CONTROL SYSTEMS	3
4	BASES for Sections 3/4.2 – POWER DISTRIBUTION LIMITS	2
5	BASES for Sections 3/4.3 – INSTRUMENTATION	2
6	BASES for Sections 3/4.4 – REACTOR COOLANT SYSTEM	7
7	BASES for Sections 3/4.5 – EMERGENCY CORE COOLING SYSTEMS (ECCS)	1
8	BASES for Sections 3/4.6 – CONTAINMENT SYSTEMS	8
9	BASES for Sections 3/4.7 – PLANT SYSTEMS	4
10	BASES for Sections 3/4.8 – ELECTRICAL POWER SYSTEMS	3
11	BASES for Sections 3/4.9 – REFUELING OPERATIONS	4
12	BASES for Sections 3/4.10 – SPECIAL TEST EXCEPTIONS	0
13	BASES for Sections 3/4.11 – RADIOACTIVE EFFLUENTS	0

END OF APPENDIX B

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 1 OF ADM-25.04 SAFETY RELATED	Section No. 2.0 Attachment No. 1 Current Revision No. 4 Effective Date 02/01/05

Title:

SAFETY LIMITS AND LIMITING SAFETY SETTINGS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 4 - Incorporated PCR 05-0059 for PCM 04078 and Tech Spec Amendment No. 138 NRC Letter dated 01/31/05 regarding WCAP-9272 Reload Methodology and Implementing 30% SG Tube Plugging Limit. (George Madden, 01/27/05)

Revision 3 - Incorporated PCR 03-1731 to change pressure to steam generator and reflect technical specification setpoint value. (Edgard Hernandez, 07/18/03)

Revision 2 – Incorporated PCR 03-1249 to revise Section 2.1.1, Figure B2.1-1 and Section 2.2.1 in accordance with Tech Spec Amendment 131; LAR 2002-06; NRC letter dated 4/18/03 regarding reduction in minimum RCS flow. (M. DiMarco, 05/02/03)

Revision 1 – Modified to reflect use of the ABB-NV critical heat flux correlation in satisfying the departure from nucleate boiling reactor core safety limit approved by License Amendment No. 118. (M. DiMarco, 11/08/01)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
4	01/27/04	G. L. Johnston Plant General Manager	01/27/05

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BASES FOR SECTION 2.0

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 or ABB-NV correlation. The CE-1 and ABB-NV DNB correlations have been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the appropriate correlation limit for DNB-SAFDL in conjunction with the Extended Statistical Combination of Uncertainties (ESCU) or the revised Thermal Design Procedure (RTDP). This value is derived through a statistical combination of the system parameter probability distribution functions with the CE-1 or ABB-NV DNB correlation uncertainties. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions. /R4

2.1 SAFETY LIMITS (continued)

BASES (continued)

2.1.1 REACTOR CORE (continued)

The curves of Figure 2.1-1 show conservative loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the DNB-SAFDL is not violated based on the ABB-NV CHF correlation for the reference 1.55 Chopped Cosine Axial Shape and Design Limit F_r^T limit shown in Figure B 2.1-1. The dashed line is not a safety limit; however, operation above this line is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 107% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe transient condition is below and to the left of these lines. /R4

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Thermal Margin/Low Pressure and Local Power Density Trip Systems, in conjunction with Limiting Conditions for Operation, the Variable Overpower Trip and the Power Dependent Insertion Limits, assure that the Specified Acceptable Fuel Design Limits on DNB and Fuel Centerline Melt are not exceeded during normal operation and design basis Anticipated Operational Occurrences. Specific verification of the DNB-SAFDL limit using an appropriate DNB correlation ensures that the reactor core safety limit is satisfied.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

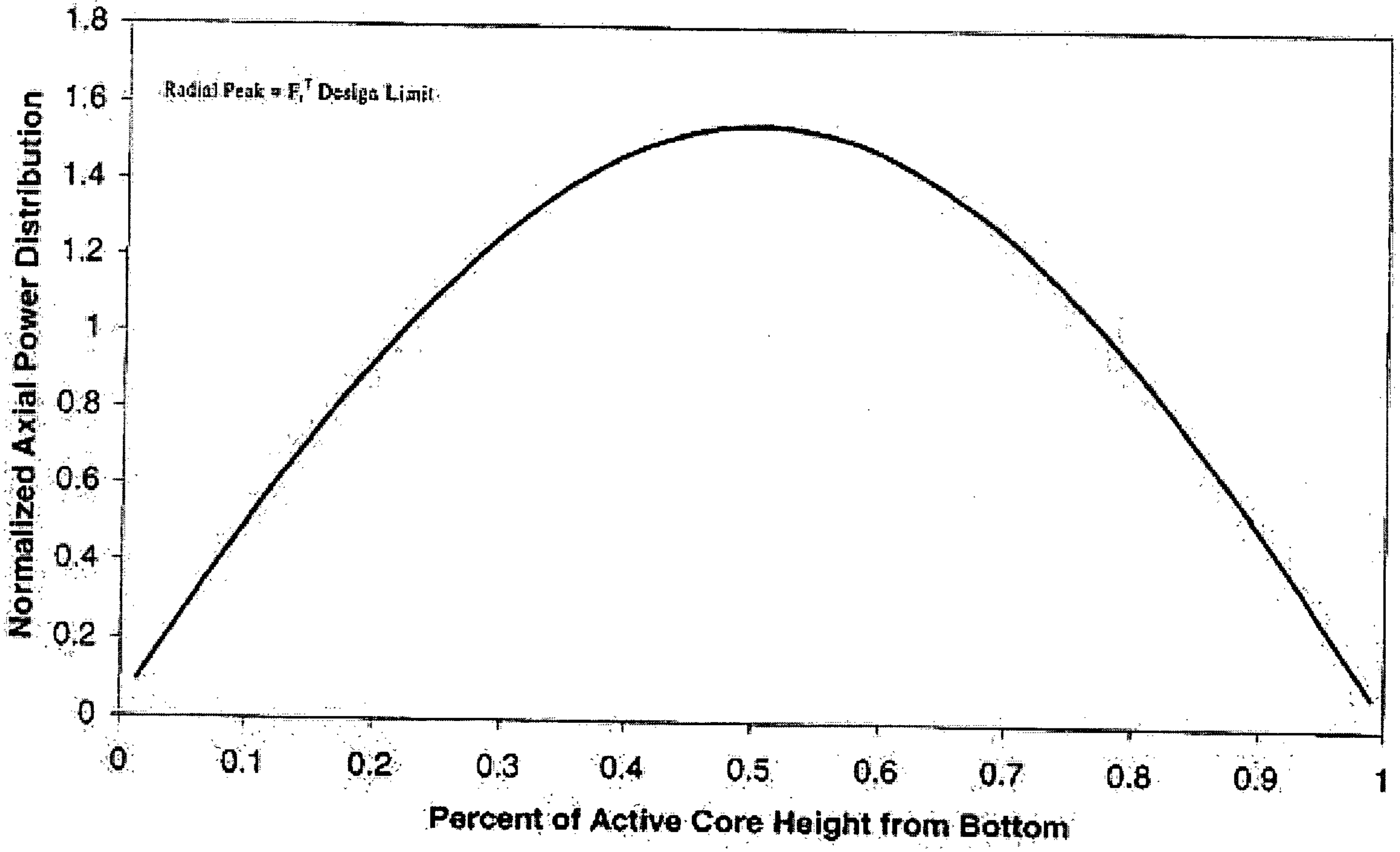
The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1971 Edition including Addenda to the Summer, 1973, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

FIGURE B 2.1-1
AXIAL POWER DISTRIBUTIONS FOR THERMAL MARGIN SAFETY LIMITS

Figure B 2.1-1
AXIAL POWER DSITRIBUTION FOR THERMAL MARGIN SAFETY LIMITS



/R4

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Power Level-High

A Reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions which are too rapid to be protected by a Pressurizer Pressure – High or Thermal Margin/Low Pressure Trip.

The Variable Power Level High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15.0% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is higher than 107% of RATED THERMAL POWER, which is the value used in the safety analysis.

2.2 LIMITING SAFETY SYSTEM SETTINGS (continued)

BASES (continued)

2.2.1 REACTOR TRIP SETPOINTS (continued)

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam line safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss without reactor trip. This trip's setpoint is at less than or equal to 2375 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the appropriate correlation limit for DNB-SAFDL, in conjunction with ESCU methodology.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1900 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time, measurement uncertainties and processing error. The allowances include: a variable (power dependent) allowance to compensate for potential power measurement error, an allowance to compensate for potential temperature measurement uncertainty; an allowance to compensate for pressure measurement error; and an allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

2.2 LIMITING SAFETY SYSTEM SETTINGS (continued)

BASES (continued)

2.2.1 REACTOR TRIP SETPOINTS (continued)

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to or concurrently with a safety injection (SIAS). This also provides assurance that a reactor trip is initiated prior to or concurrently with an MSIS.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 626 psia is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of 30 psi in the safety analyses.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide sufficient time for any operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost. This trip also protects against violation of the specified acceptable fuel design limits (SAFDL) for DNBR, offsite dose and the loss of shutdown margin for asymmetric steam generator transients such as the opening of a main steam safety valve or atmospheric dump valve.

2.2 LIMITING SAFETY SYSTEM SETTINGS (continued)

BASES (continued)

2.2.1 REACTOR TRIP SETPOINTS (continued)

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower excore neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15% power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

RCP Loss of Component Cooling Water

A loss of component cooling water to the reactor coolant pumps causes a delayed reactor trip. This trip provides protection to the reactor coolant pumps by ensuring that plant operation is not continued without cooling water available. The trip is delayed 10 minutes following a reduction in flow to below the trip setpoint and the trip does not occur if flow is restored before 10 minutes elapses. No credit was taken for this trip in the safety analysis. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protective System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. The trip is not credited in any design basis accident evaluated in UFSAR Chapter 15; however, the trip is considered in the safety analysis in that the presence of this trip function precluded the need for specific analyses of other events initiated from subcritical conditions.

2.2 LIMITING SAFETY SYSTEM SETTINGS (continued)

BASES (continued)

2.2.1 REACTOR TRIP SETPOINTS (continued)

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection against DNB in the event of a sudden significant decrease in RCS flow. The Reactor trip setpoint on low RCS flow is calculated by a relationship between steam generator differential pressure, core inlet temperature, instrument errors and response times. When the calculated RCS flow falls below the trip setpoint in an automatic reactor trip signal is initiated. The trip setpoint and allowable values ensure that for a degradation of RCS flow resulting from expected transients, a reactor trip occurs to prevent violation of local power density or DNBR safety limits.

Loss of Load (Turbine)

The Loss of Load (Turbine) trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip is an equipment protective trip only and is not required for plant safety. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Asymmetric Steam Generator Transient Protective Trip Function (ASGTPTF)

The ASGTPTF utilizes steam generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those Anticipated Operational Occurrences associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single Main Steam Isolation Valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 2 OF ADM-25.04 SAFETY RELATED	Sections No. **3.0 & 4.0** Attachment No. **2** Current Revision No. **2** Effective Date **04/29/09**

Title:

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 2 - Incorporated PCR 09-1217 for CR 2009-4976 to incorporate TSTF-434 - Overlap Testing - in Bases for SR 4.0.1. (Author: Ken Frehafer)

Revision 1 – Updated TS Bases for TS Amendment No. 129 - missed surveillances. (Larry Donghia, 01/03/03)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
2	04/23/09	C. Costanzo Plant General Manager	04/23/09

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BASES FOR SECTIONS 3.0 & 4.0

3/4.0 APPLICABILITY

BASES

The specifications of this section establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

3.0.1 This specification establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

3/4.0 APPLICABILITY (continued)

BASES (continued)

3.01 (continued)

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

3.0.2 This specification establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

3/4.0 APPLICABILITY (continued)

BASES (continued)

3.0.3 This specification establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically address by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

3/4.0 APPLICABILITY (continued)

BASES (continued)

3.03 (continued)

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

3/4.0 APPLICABILITY (continued)

BASES (continued)

3.0.4 This specification establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with the ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

The specifications of this section establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

3/4.0 APPLICABILITY (continued)

BASES (continued)

4.0.1 SR 4.0.1 establishes the requirement that Surveillance Requirements (SR) must be met during the MODES or other specified conditions in the applicability for which the requirements of the Limiting Condition for Operation apply, unless otherwise specified in the individual SRs. This Specification is to ensure that SRs are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a SR within the specified frequency, in accordance with SR 4.0.2, constitutes a failure to meet a Limiting Condition for Operation (except as allowed by SR 4.0.3). Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps. /R2

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. the systems or components are known to be inoperable, although still meeting the SRs, or

b. the requirements of the SR(s) are known to be not met between required SR performances.

SRs do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated Limiting Condition for Operation are not applicable, unless otherwise specified. The SRs associated with a SPECIAL TEST EXCEPTION (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

SRs, including SRs invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. SRs have to be met and performed in accordance with SR 4.0.2, prior to returning equipment to OPERABLE status.

3/4.0 APPLICABILITY (continued)

BASES (continued)

4.01 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable SRs are not failed and their most recent performance is in accordance with SR 4.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process follow.

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

3/4.0 APPLICABILITY (continued)

BASES (continued)

4.0.2 This specification establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified within an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

4.0.3 SR 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a SR has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the SR has not been performed in accordance with SR 4.0.2, and not at the time that the specified frequency was not met.

This delay period provides adequate time to complete SRs that have been missed. This delay period permits the completion of a SRs requirement before complying with required ACTION(s) or other remedial measures that might preclude completion of the SR.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the SR, the safety significance of the delay in completing the required SR, and the recognition that the most probable result of any particular SR being performed is the verification of conformance with the requirements.

3/4.0 APPLICABILITY (continued)

BASES (continued)

4.03 (continued)

When a SR with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 4.0.3 allows for the full delay period of up to the specified frequency to perform the SR. However, since there is not a time interval specified, the missed SR should be performed at the first reasonable opportunity.

SR 4.0.3 provides a time limit for, and allowances for the performance of, a SR that becomes applicable as a consequence of MODE changes imposed by required ACTION(s).

Failure to comply with the specified frequency for a SR is expected to be an infrequent occurrence. Use of the delay period established by SR 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed SR will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the SR) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the SR. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, *Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants*. This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed SRs for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the course of action. All cases of a missed SR will be placed in the licensee's Corrective Action Program.

3/4.0 APPLICABILITY (continued)

BASES (continued)

4.03 (continued)

If a SR is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the completion times of the required ACTION(s) for the applicable Limiting Condition for Operation begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the completion times of the required ACTION(s) for the applicable Limiting Condition for Operation begin immediately upon the failure of the surveillance.

Completion of the SR within the delay period allowed by this specification, or within the completion time of the ACTIONS, restores compliance with SR 4.0.1.

4.0.4 This specification establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition or operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not part of these Technical Specifications.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 3 OF ADM-25.04 SAFETY RELATED	Section No. **3/4.1** Attachment No. **3** Current Revision No. **3** Effective Date **05/29/06**

Title:

REACTIVITY CONTROL SYSTEMS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 3 - Incorporated PCR 06-1727 for PCM 05197, CR 2006-15180 to update reactivity controls and RCS bases, and make corrections per CR. (Ken Frehafer, 05/25/06)

Revision 2 – Incorporated PCR 04-3132 to implement amendments 194/136. (K. W. Frehafer, 11/24/04)

Revision 1 – Changes made to reflect TS Amendment #122. (K.W. Frehafer, 11/30/01)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
Revision	FRG Review Date	Approved By	Approval Date
3	05/25/06	C. Costanzo Plant General Manager	05/25/06

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BASES FOR SECTION 3/4.1

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN as specified in the COLR for Specification 3.1.1.1 is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. At earlier times in core life, the minimum SHUTDOWN MARGIN required for the most restrictive conditions is less than that at EOL. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from any postulated accident are minimal and a SHUTDOWN MARGIN as specified in the COLR for Specification 3.1.1.2 provides adequate protection.

3/4.1.1.3 BORATION DILUTION

A minimum flow rate of at least 3000 gpm provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 gpm will circulate an equivalent Reactor Coolant System volume of 10,931 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.1 BORATION CONTROL (continued)

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of the limit specified in the COLR after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions. This requirement can be met for a range of boric acid concentrations in the Boric Acid Makeup Tank (BAMT) and Refueling Water Tank (RWT). This range is bounded by 5350 gallons of 3.5 weight percent (6119 ppm boron) from the BAMT and 16,000 gallons of 1720 ppm borated water from the RWT to 8650 gallons of 2.5 weight percent (4371 ppm boron) boric acid from BAMT and 12,000 gallons of 1720 ppm borated water from the RWT. A minimum of 35,000 gallons of 1720 ppm boron is required from the RWT if it is to be used to borate the RCS alone.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.2 BORATION SYSTEMS (continued)

Temperature changes in the RCS impose reactivity changes by means of the moderator temperature coefficient. Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. Small changes in RCS temperature are unavoidable and so long as the required SDM is maintained during these changes, any positive reactivity additions will be limited to acceptable levels. Introduction of temperature changes must be evaluated to ensure they do not result in a loss of required SDM.

The boron capability required below 200°F is based upon providing a SHUTDOWN MARGIN corresponding to its COLR limit after xenon decay and cooldown from 200°F to 140°F. This condition requires either 6750 gallons of 1720 ppm – 2100 ppm borated water from the refueling water tank or 3550 gallons of 2.5 to 3.5 weight percent boric acid solution from the boric acid makeup tanks.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

Ensuring that the BAM pump discharge pressure is met satisfies the periodic surveillance requirement to detect gross degradation caused by impeller structural damage or other hydraulic component problems. Along with this requirement, Section XI of the ASME Code verifies the pump developed head at one point on the pump characteristic curve to verify both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. Surveillance Requirements are specified in the In-service Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 15 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 63-minute time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 63-minute time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution. /R3

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (continued)

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

The requirement to reduce power in certain time limits depending upon the previous F_r^T is to eliminate a potential nonconservatism for situations when a CEA has been declared inoperable. A worst-case analysis has shown that a DNBR SAFDL violation may occur after the CEA misalignment if this time requirement is not met. This potential DNBR SAFDL violation is eliminated by limiting the time operation is permitted at full power before power reductions are required. These reductions will be necessary once the deviated CEA has been declared inoperable. This time allowed to continued operation at a reduced power level can be permitted for the following reasons: /R3

1. The margin calculations that support the Technical Specifications are based on a steady-state radial peak of F_r^T = the limits of Specification 3.2.3.

2. When the actual F_r^T < the limits of Specification 3.2.3, significant additional margin exists.

3. This additional margin can be credited to offset the increase in F_r^T with time that can occur following a CEA misalignment.

4. This increase in F_r^T is caused by xenon redistribution.

5. The present analysis can support allowing a misalignment to exist without correction, if the time constraints and initial F_r^T limits of COLR Figure 3.1-1a are met. /R3

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (continued)

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 515°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Long Term Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Power Dependent Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long-term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 4 OF ADM-25.04 SAFETY RELATED	Section No. **3/4.2** Attachment No. **4** Current Revision No. **2** Effective Date **02/01/05**

Title:

POWER DISTRIBUTION LIMITS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 2 - Incorporated PCR 05-0059 for PCM 04078 and Tech Spec Amendment No. 138 NRC Letter dated 01/31/05 regarding WCAP-9272 Reload Methodology and Implementing 30% SG Tube Plugging Limit. (George Madden, 01/27/05)

Revision 1 – Incorporated PCR 03-1249 to revise Section 3/4.2.5 in accordance with Tech Spec Amendment 131; LAR 2002-06; NRC letter dated 4/18/03 regarding reduction in minimum RCS flow. (M. DiMarco, 05/02/03)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
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BASES FOR SECTION 3/4.2

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: (1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, (2) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and (3) the measured linear heat rate obtained from a previous power distribution map using incore detectors meets the criteria of Specification 3.2.1.

Although linear heat rate is continuously monitored when using the Incore Detector Monitoring System, the formal measurement of $LHR^M(z)$ is normally made under steady state conditions. Should the Incore Detector Monitoring System become inoperable, the last measurement of linear heat rate, $LHR^M(z)$, would remain applicable, but only under steady state conditions. With the Incore Detector Monitoring System inoperable, and using only the Excore Detector Monitoring System, variations in power distributions resulting from normal operation maneuvers cannot be directly monitored. Variations from the steady state power distribution are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, z, is called W(z).

To account for power distribution transients encountered during normal operation, the transient limits for LHR(z) are established utilizing the cycle dependent function W(z).

$LHR^M(z)$ is the measured LHR(z) increased by the allowances for manufacturing tolerances and calorimetric uncertainty.

/R2

3/4.2 POWER DISTRIBUTION LIMITS (continued)

BASES (continued)

The W(z) table is provided in the COLR for discrete core elevations. LHR(z) evaluations for comparison to the transient limits are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and
b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

If the two most recent LHR(z) evaluations show an increase in the quantity:

[LHR^M(z)] normalized to 100% RATED THERMAL POWER

it is not guaranteed that LHR(z) will remain within the transient limit during the following surveillance interval. Therefore, LHR(z) is increased by the penalty factor specified in the COLR and compared to the transient LHR(z) limit.

If the relationship:

$$LHR^M(z) \leq \frac{LHR}{W(z)}$$

is not satisfied, comply with the requirements of Specification 3.2.1 for LHR^M(z) exceeding its limit.

Reduce THERMAL POWER at least 1% for each 1% LHR(z) exceeds the limit after each determination of LHR(z).

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of COLR Figure 3.2-1. The setpoints for these alarms include allowances, set in conservative directions, for (1) a measurement-calculational uncertainty factor, (2) an engineering uncertainty factor, (3) an allowance for axial fuel densification and thermal expansion, and (4) a THERMAL POWER measurement uncertainty factor.

/R2

3/4.2 POWER DISTRIBUTION LIMITS (continued)

BASES (continued)

3/4.2.3 and 3/4.2.4 TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitation on T_q is provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_r^T and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, the Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density – High LCOs and LSSS setpoints remain valid. /R2

An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The requirement that the measured value of T_q be multiplied by the calculated values of F_r to determine F_r^T is applicable only when F_r is calculated with a non-full core power distribution analysis code. When monitoring a reactor core power distribution, F_r with a full core power distribution analysis code the azimuthal tilt is explicitly accounted for as part of the radial power distribution used to calculate F_r. /R2

The Surveillance Requirements for verifying that F_r^T and T_q are within their limits provide assurance that the actual values of F_r and T_q do not exceed the assumed values. Verifying F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded. /R2

3/4.2 POWER DISTRIBUTION LIMITS (continued)

BASES (continued)

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the appropriate correlation limit for DNB-SAFDL in conjunction with ESCU or RTDP methodology throughout each analyzed transient. /R2

These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow based on maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient. /R2

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 5 OF ADM-25.04 SAFETY RELATED	Section No. 3/4.3 Attachment No. 5 Current Revision No. 2 Effective Date 06/27/09

Title:

INSTRUMENTATION

Responsible Department: **Licensing**

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BASES FOR SECTION 3/4.3

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

CE Owners Group topical report CEN-403, Revision 1-A, March 1996, provides the basis to allow ESFAS subgroup relay testing on a STAGGERED TEST BASIS. Such testing requires each subgroup relay to be tested at least once per 18 months (refueling cycle), with approximately equal numbers of relays being tested at 6 month subintervals. Subgroup relays which cannot be tested with the unit at power should be scheduled for testing during plant shutdowns. If two or more ESFAS subgroup relays fail in a 12-month period, the design, maintenance, and testing of all ESFAS subgroup relays should be considered to evaluate the adequacy of the surveillance interval. If it is determined that the surveillance interval is inadequate for detecting a single relay failure, the surveillance interval should be decreased such that an ESFAS subgroup relay failure prior to occurrence of a second failure can be detected.

3/4.3 INSTRUMENTATION (continued)

BASES (continued)

3/4.3.1 and 3/4.3.2 (continued)

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, provided that such tests demonstrate total channel response time as defined. CEOG Topical Report CE NPSD-1167, and FPL No Significant Hazards Evaluation PSL-ENG-SEIS-03-043 provide the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in these documents. The allocated sensor response time must be verified prior to placing a new component in operation and re-verified after maintenance that may adversely affect the sensor response time (e.g., replacement of a transmitter DP cell or variable damping circuits). Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The CEOG topical report and FPL evaluation only cover certain sensor model numbers. If sensors are replaced with types not previously evaluated, then periodic response time testing (RTT) for the new sensor must either be performed and the appropriate changes made to plant procedures, or an additional request for RTT elimination must be submitted and approved by the NRC. If, however, the replacement sensor is one for which RTT elimination has been approved, then FPL may modify the plant procedures, using an allocated response time based upon a vendor-supplied response time value, or upon statistical analysis of historical data for that transmitter type and model.

The Safety Injection Actuation Signal (SIAS) provides direct actuation of the Containment Isolation Signal (CIS) to ensure containment isolation in the event of a small break LOCA.

3/4.3 INSTRUMENTATION (continued)

BASES (continued)

3/4.3.3 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. /R2

Surveillance Requirement 4.3.3.2 ensures that the channel actuation response times are less than the maximum times assumed in the analyses. Testing of the final actuating devices, which make up the bulk of the response time, is included in the surveillance testing. /R2

3/4.3.5 REMOTE SHUTDOWN INSTRUMENTATION /R2

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown system instrumentation ensures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control circuits, and transfer switches are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3 INSTRUMENTATION (continued)

BASES (continued)

3/4.3.6 ACCIDENT MONITORING INSTRUMENTATION /R2

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.7 DELETED /R2

3/4.3.8 DELETED /R2

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FPL	**ST. LUCIE UNIT 2** **TECHNICAL SPECIFICATIONS BASES ATTACHMENT 6 OF ADM-25.04** SAFETY RELATED **INFORMATION USE**	Section No. **3/4.4** Attachment No. **6** Current Revision No. **7** Effective Date **06/27/09**

Title:

REACTOR COOLANT SYSTEM

Responsible Department: **Licensing**

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Revision 6 - Incorporated PCR 09-0379 to update Unit 2 TS Bases to reflect 55 EFPY PT Curve implementation. (Author: Ken Frehafer)

Revision 5 – Incorporated PCR 07-2483 to implement TS Bases (3/4.4 Attachment 6) associated with the TSTF-449 SG Tube Integrity Program. (K.W. Frehafer, 08/16/07)

Revision 4 - Incorporated PCR 06-1935 for PCM 05197 to update Unit 2 tech spec bases sections 3/4.4 and 3/4.7. (Modesto Jimenez, 06/28/06)

Revision 3 - Incorporated PCR 06-1727 for PCM 05197, CR 2006-15180 to update reactivity controls and RCS bases, and make corrections per CR. (Ken Frehafer, 05/25/06)

Revision 2 – Incorporated PCR 03-0626 for PM 02-09-030 and PSL-ENG-SENS-02-044 R0 to include update from PCM 03021, Att. 4.9, R0, which updates Table B3/4.4-1, Reactor Vessel Toughness." (C.J. Wasik, 04/18/03)

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

If no coolant loops are in operation during shutdown operations, suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1.1 or 3.1.1.2 is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (continued)

The restriction on starting a reactor coolant pump in MODES 4 and 5, with two idle loops and one or more RCS cold leg temperatures less than or equal to that specified in Table 3.4-3 is provided to prevent RCS pressure transients, caused by energy additions from the secondary system from exceeding the limits of Appendix G to 10 CFR 50. The RCS will be protected against overpressure transients by (1) sizing each PORV to mitigate the pressure transient of an inadvertent safety injection actuation in a water-solid RCS with pressurizer heaters energized, (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 40°F above each of the RCS cold leg temperatures, (3) using SDCRVs to mitigate RCP start transients and the transients caused by inadvertent SIAS actuation and charging water, and (4) rendering one HPSI pump inoperable when the RCS is at low temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 212,182 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power-operated relief valve or steam dump valves.

3/4.4 REACTOR COOLANT SYSTEM (continued)

<u>**BASES**</u> (continued)

3/4.4.2 SAFETY VALVES (continued)

Surveillance Requirements are specified in the Inservice Testing Program. Pressurizer code safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code, which provides the activities and the frequency necessary to satisfy the Surveillance Requirements. No additional requirements are specified.

The pressurizer code safety valve as-found setpoint is 2500 psia +/- 2% for OPERABILITY; however, the valves are reset to 2500 psia +/- 1% during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

3/4.4.3 PRESSURIZER

A OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.4 PORV BLOCK VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs in conjunction with a reactor trip on a Pressurizer Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2, or 3.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Since it is impractical and undesirable to actually open the PORVs to demonstrate their reclosing, it becomes necessary to verify OPERABILITY of the PORV block valves to ensure capability to isolate a malfunctioning PORV. As the PORVs are pilot operated and require some system pressure to operate, it is impractical to test them with the block valve closed.

The PORVs are sized to provide low temperature overpressure protection (LTOP). Since both PORVs must be OPERABLE when used for LTOP, both block valves will be open during operation with the LTOP range. As the PORV capacity required to perform the LTOP function is excessive for operation in MODE 1, 2, or 3, it is necessary that the operation of more than one PORV be precluded during these MODES. Thus, one block valve must be shut during MODES 1, 2, and 3.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "Reactor Coolant Loops and Coolant Circulation, Startup and Power Operation," LCO 3.4.1.2, "Hot Standby," LCO 3.4.1.3, "Hot Shutdown," LCO 3.4.1.4.1, "Cold Shutdown - Loops Filled," and LCO 3.4.1.4.2, "Cold Shutdown - Loops Not Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

SG tubing is subject to a variety of degradation mechanism. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.4.l, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.l, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.l. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Background (continued)

Specification 6.8.4.l has two parts to address the replacement SG and original SG designs. Specification 6.8.4.l.1. applies to the replacement SG design. TS 6.8.4.l.2 applies to the original SGs and contains requirements such as a sleeving repair method, alternate repair criteria and additional inspection requirements, which apply only to the original SG design and can be removed following SG replacement.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes that contaminated secondary fluid is released via the main steam safety valves and/or atmospheric dump valves. The majority of the activity released to the atmosphere results from the tube rupture.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 0.5 gpm total and 0.25 gpm through any one SG or is assumed to increase to 0.5 gpm total through all SGs and 0.25 gpm through any one SG as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2) and the requirements of 10 CFR 50.67 (Ref. 7). /R7

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still have tube integrity. Tube repair (i.e., sleeving) is applicable only to the original SGs.

In the context of this Specification, a SG tube for the replacement SGs is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For the original SGs, when the alternate repair criteria in TS Section 6.8.4.l.2.c.4 are applied a SG tube is defined as the length of the tube, including the tube wall and any repairs made to it, between 10.3 inches below the bottom of the hot leg expansion transition or top of the tubesheet (whichever is lower) and the tube-to-tubesheet weld at the tube outlet. If a portion of a tube sleeve extends below 10.3 inches from the bottom of the hot leg expansion transition or the top of the tubesheet (whichever is lower) a SG tube is defined as the length of the tube between the bottom of the sleeve to the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.l., "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Limiting Condition for Operation (LCO) (continued)

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 0.5 gpm total and 0.25 gpm through any one SG. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident. /R7

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Limiting Condition for Operation (LCO) (continued)

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "Reactor Coolant System operational leakage," and limits primary-to-secondary leakage through any one SG to 150 gpd at room temperature. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Applicability

SG tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in POWER OPERATION, STARTUP, HOT STANDBY and HOT SHUTDOWN.

RCS conditions are far less challenging in COLD SHUTDOWN and REFUELING than during POWER OPERATION, STARTUP, HOT STANDBY and HOT SHUTDOWN. In COLD SHUTDOWN and REFUELING, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

The ACTIONS are modified by a Note clarifying that the CONDITIONS may be entered independently for each SG tube. This is acceptable because the required ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the required ACTIONS may allow for continued operation, and subsequently affected SG tubes are governed by subsequent Condition entry and application of associated required ACTIONS.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

ACTIONS (continued)

a.1 and a.2

ACTIONS a.1 and a.2 apply if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by Surveillance Requirement (SR) 4.4.5.2. Tube repair (i.e., sleeving) is applicable only to the original SGs. An evaluation of SG tube integrity of the affected tube(s) must be made. SG tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged or repaired has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, ACTION b applies.

An allowable completion time of seven days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, ACTION a.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged or repaired prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This allowable completion time is acceptable since operation until the next inspection is supported by the operational assessment.

3/4.4 REACTOR COOLANT SYSTEM (continued)

<u>**BASES**</u> (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

<u>ACTIONS</u> (continued)

b.

If the requirements and associated completion time of ACTION a are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. The allowable completion times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

SR 4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, "Steam Generator Program Guidelines" (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Surveillance Requirements (continued)

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.l contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.4.5.2 During a SG inspection any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.l are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program (Specification 6.8.4.l.2.). Tube repair (i.e., sleeving) is applicable only to original SGs.

The frequency of prior to entering HOT SHUTDOWN following a SG tube inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

3/4.4 REACTOR COOLANT SYSTEM (continued)

<u>BASES</u> (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. 10 CFR 50 Appendix A, GDC 19

3. Deleted /R7

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB

5. Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976

6. EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines"

7. 10 CFR 50.67

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. The LCO is consistent with NUREG-1432, Revision 1, and is satisfied when leakage detection monitors of diverse measurement means are OPERABLE in MODES 1, 2, 3, and 4. Monitoring the reactor cavity sump inlet flow rate, in combination with monitoring the containment particulate or gaseous radioactivity, provides an acceptable minimum to assure that unidentified leakage is detected in time to allow actions to place the plant in a safe condition when such leakage indicates possible pressure boundary degradation.

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the sources of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Background (continued)

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

Applicable Safety Analyses

The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary leakage from all steam generators (SGs) is 0.5 gpm total through all SGs and 0.25 gpm through any one SG or is assumed to increase to 0.5 gpm total through all SGs and 0.25 gpm through any one SG as a result of accident induced conditions. The LCO requirements to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd is based on room temperature conditions. When this value is adjusted for operating conditions, it is less than the leakage limit of 0.25 gpm (measured at operating temperature) through any one SG assumed in the accident analysis. /R7

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Applicable Safety Analyses (continued)

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released mainly via the safety valves or atmospheric dump valves and only briefly steamed to the condenser. The 0.5 gpm total through all SGs and 0.25 gpm through any one SG primary to secondary leakage safety analysis assumption is relatively inconsequential. /R7

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes a value of 0.25 gpm primary to secondary leakage through each generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in GDC 19 and the requirements of 10 CFR 50.67. /R7

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Limiting Condition for Operation (LCO) (continued)

b. UNIDENTIFIED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Primary-to-Secondary Leakage Through Any One Steam Generator

The limit of 150 gpm per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion is conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically know and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Limiting Condition for Operation (LCO) (continued)

Reactor Coolant System Pressure Isolation Valve Leakage

Leakage is measured through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS Leakage when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

Applicability

In POWER OPERATION, STARTUP, HOT STANDBY and HOT SHUTDOWN, the potential for PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

ACTIONS

a.

If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to HOT STANDBY with 6 hours and COLD SHUTDOWN within the following 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

b.

UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE is excess of the LCO limits must be reduced to within the limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. Otherwise, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. This ACTION is necessary to prevent further deterioration of the Reactor Coolant Pressure Boundary.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

ACTIONS (continued)

c.

The leakage from any RCS Pressure Isolation Valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two manual or deactivated automatic valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. With one or more RCS Pressure Isolation Valves with leakage greater than that allowed by Specification 3.4.6.2.e, within 4 hours, at least two valves in each high pressure line having a non-functional valve must be closed and remain closed to isolate the affected line(s). In addition, the ACTION statement for the affected system must be followed and the leakage from the remaining Pressure Isolation Valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1 shall be recorded daily. If these requirements are not met, the reactor must be brought to at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

d.

With RCS leakage alarmed and confirmed in a flow path with no flow indication, commencement of an RCS water inventory balance is required within 1 hour to determine the leak rate. This action is not applicable to primary-to-secondary leakage.

The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the Reactor Coolant Pressure Boundary are much lower, and further deterioration is much less likely.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Surveillance Requirements

4.4.6.2.1

Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

a. and b.

These SRs demonstrate that the RCS operational leakage is within the LCO limits by monitoring the containment atmosphere gaseous and particulate radioactivity monitor and the containment sump level and discharge at least once per 12 hours.

c.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows). The Surveillance is modified by a note that states that this Surveillance Requirement is not required to be performed until 12 hours after establishment of steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Surveillance Requirements (continued)

4.4.6.2.1 (continued)

c. (continued)

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity, containment normal sump inventory and discharge, and reactor head flange leakoff. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. The reactor cavity (containment) sump and containment atmosphere radioactivity leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System Leakage Detection Systems." /R7

The note also states that this SR is not applicable to primary-to secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

This 72-hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

d.

This SR demonstrates that the RCS operational leakage is within the LCO limits by monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Surveillance Requirements (continued)

4.4.6.2.1 (continued)

e.

This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity" should be evaluated. The 150 gpd limit is measured at room temperature as described in Reference 5. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator. /R7

The Surveillance Requirement is modified by a note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Surveillance Requirements (continued)

4.4.6.2.2

a. through d.

This Surveillance Requirement verifies RCS Pressure Isolation Valve check valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation check valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

4.4.6.2.3

a. and b.

This Surveillance Requirement verifies RCS Pressure Isolation Valve motor-operated valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation motor-operated valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

References

1. 10 CFR 50, Appendix A, GDC 30
2. Regulatory Guide 1.45
3. UFSAR, Section 15.6.3
4. NEI 97-06, "Steam Generator Program Guidelines"
5. EPRI "PWR Primary-to-Secondary Leak Guidelines"

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed 10 CFR 50.67 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 0.5 gpm total primary-to-secondary leakage through all SGs and 0.25 gpm through any one SG, and a loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation. /R7

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take correction action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. /R7

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron indication damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degrees F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at 55 EFPY, and they include adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature can be predicated using a) the initial RT_{NDT}, b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift as recommended by Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or other approved method. The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at 55 EFPY.

The actual shift in RT_{NDT} of the vessel materials will be benchmarked periodically during operation, by removing and evaluating, in accordance with 10 CFR 50 Appendix H and ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and the vessel inside radius are essentially identical, the measured transition temperature shift in RT_{NDT} for a set of material samples can be compared to the predications of RT_{NDT} that were used for preparations of the pressure/temperature limits curves. If the measured delta RT_{NDT} values from the surveillance capsule are not conservatively within the measurement uncertainty of the prediction method, then heat up and cooldown curves must be re-evaluated.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements for Appendix G to 10 CFR 50.

The maximum RT_{NDT} all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maxi-mum of 20% of the system's hydrostatic test pressure.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two SDCRVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold leg temperatures are less than or equal to the LTOP temperatures. The Low Temperature Overpressure Protection System has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 40°F above the RCS cold leg temperatures with the pressurizer water-solid.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch RT (°F) NDT @ 50 ft-lb	Minimum Upper Shelf Cv energy for Transverse Direction Charpy(1) Ft-lb
122-102A	M-604-1	SA 533B C1 1	Upper Shell Plate	0	+50	---
122-102B	M-604-2	SA 533B C1 1	Upper Shell Plate	+10	+50	---
122-102C	M-604-3	SA 533B C1 1	Upper Shell Plate	-10	+10	---
124-102B	M-605-1	SA 533B C1 1	Immediate Shell Plate	0	+30	105
124-102C	M-605-2	SA 533B C1 1	Immediate Shell Plate	-10	+10	113
124-102A	M-605-3	SA 533B C1 1	Immediate Shell Plate	-20	0	113
142-102C	M-4116-1	SA 533B C1 1	Lower Shell Plate	-30	+20	91
142-102B	M-4116-2	SA 533B C1 1	Lower Shell Plate	-50	+20	105
142-102A	M-4116-3	SA 533B C1 1	Lower Shell Plate	-40	+20	100
102-101	M-4110-1	SA 533B C1 1	Closure Head	-10	+30	---
106-101	M-4101-1	SA 508 C1 2	Closure Head Flange	0	0	---
128-101A	M-4102-1	SA 508 C1 2	Inlet Nozzle	-20	-20	---
128-101D	M-4102-2	SA 508 C1 2	Inlet Nozzle	-20	-20	---
128-101B	M-4102-3	SA 508 C1 2	Inlet Nozzle	0	0	---
128-101C	M-4102-4	SA 508 C1 2	Inlet Nozzle	-10	-10	---
128-301B	M-4103-1	SA 508 C1 2	Outlet Nozzle	-20	-20	---
128-301A	M-4103-2	SA 508 C1 2	Outlet Nozzle	-30	-30	---
126-101	M-602-1	SA 508 C1 2	Vessel Flange	-30	-10	---
131-102A	M-4104-1	SA 508 C1 1	Inlet Nozzle Safe End	-20	+20	---
131-102D	M-4104-2	SA 508 C1 1	Inlet Nozzle Safe End	-20	+20	---
131-102B	M-4104-3	SA 508 C1 1	Inlet Nozzle Safe End	-20	+20	---
131-102C	M-4104-4	SA 508 C1 1	Inlet Nozzle Safe End	-20	+20	---
131-101B	M-4105-1	SA 508 C1 1	Outlet Nozzle Safe End	-10	0	---
131-101A	M-4105-2	SA 508 C1 1	Outlet Nozzle Safe End	-10	0	---
152-101	M-4112-1	SA 533B C1 1	Bottom Head Dome	-50	-40	---
154-102 (A to F)	M-4111-1	SA 533B C1 1	Bottom Head Torus	-40	+40	---
104-102 (A to D)	M-4109-1	SA 533B C1 1	Closure Head Torus	-60	-10(2)	---

(1) Reported only for beltline region plates

(2) A 10°F RT_{NDT} increase shall be added to the Closure Head Torus as a result of using a temper bead weld procedure identified in PCM 03021.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. This programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a (g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1973.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 7 OF ADM-25.04 SAFETY RELATED	Section No. **3/4.5** Attachment No. **7** Current Revision No. **1** Effective Date **12/01/04**

Title:

EMERGENCY CORE COOLING SYSTEMS (ECCS)

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 1 – Incorporated PCR 04-3132 to implement amendments 194/136. (K. W. Frehafer, 11/24/04)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
1	11/23/04	G. L. Johnston Plant General Manager	11/24/04

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BASES FOR SECTION 3/4.5

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the assumptions used for safety injection tank injection in the safety analysis are met.

The safety injection tank power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limit of 72 hours for operation with an SIT that is inoperable due to boron concentration not within limits, or due to the inability to verify liquid volume or cover-pressure, considers that the volume of the SIT is still available for injection in the event of a LOCA. If one SIT is inoperable for other reasons, the SIT may be unable to perform its safety function and, based on probability risk assessment, operation in this condition is limited to 24 hours.

The practice of calibrating and testing the SIT isolation valve interlock function below 515 psia (the current plant practice is to set and test the interlock function at 500 psia) meets the requirements of Technical Specification Surveillance 4.5.1.1.d.1. The staff accepted that testing the SIT isolation interlock at a more conservative setpoint demonstrates operability at and above the setpoint (NRC letter from William C. Gleaves to J.A. Stall dated November 2, 1999, subject "St. Lucie Unit 2 – Amendment Request Regarding Safety Injection Tank and Shutdown Cooling System Isolation Interlock Surveillances (TAC No. MA5619)."

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) (continued)

<u>**BASES**</u> (continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS hot leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

TS 3.5.2, ACTION a.1. provides an allowed outage/action completion time (AOT) of up to 7 days from initial discovery of failure to meet the LCO provided the affected ECCS subsystem is inoperable only because its associated LPSI train is inoperable. This 7 day AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk-informed" AOT extension. Entry into this ACTION requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP) which is described in the Administrative Procedure (ADM-17.08) that implements the Maintenance Rule pursuant to 10 CFR 50.65.

In Mode 3 with RCS pressure < 1750 psia and in Mode 4, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) (continued)

BASES (continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (continued)

The requirement for one high pressure safety injection pump to be rendered inoperable prior to entering MODE 5, although the analysis supports actuation of safety injection in a water solid RCS with pressurizer heaters energized, provides additional administrative assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or SDCRV. A limit on the maximum number of operable HPSI pumps is not necessary when the pressurizer manway cover or the reactor vessel head is removed.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. The Surveillance Requirement for throttle valve position stops, along with appropriate post-maintenance flow balance testing,* provides assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point on the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. Surveillance Requirements are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

* Refer to UFSAR for flow balancing requirements

/R1

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) (continued)

<u>BASES</u> (continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (continued)

The practice of calibrating and testing the SDC isolation valve interlock function below 515 psia (the current plant practice is to set and test the interlock function at 500 psia) meets the requirements of Technical Specification Surveillance 4.5.2.e.1. The staff accepted that testing the SDC isolation interlock at a more conservative setpoint demonstrates operability at and above the setpoint (NRC letter from William C. Gleaves to J.A. Stall dated November 2, 1999, subject "St. Lucie Unit 2 – Amendment Request Regarding Safety Injection Tank and Shutdown Cooling System Isolation Interlock Surveillances (TAC No. MA5619)."

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the Refueling Water Tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 8 OF ADM-25.04 SAFETY RELATED	Section No. 3/4.6 Attachment No. 8 Current Revision No. 8 Effective Date 06/27/09

Title:

CONTAINMENT SYSTEMS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 8 – Incorporated PCR 08-6765 for CR 2007-32178 for Bases changes to Technical Specifications 155 for License Amendments 152 and 153. Procedure changes to implement AST were reviewed in ORG 07-041 on 5/29/07 as part of the license amendment submittal. (Author: Ken Frehafer)

Revision 7 - Incorporated PCR 08-0488 for CR 2007-5341 to implement TS amendments 204/151 - elimination of H2 Recombiner TS and relocation of H2 Analyzer TS to licensee controlled document (UFSAR). (K.W. Frehafer, 03/27/08)

Revision 6 - Incorporated PCR 04-3132 to implement amendments 194/136. (K. W. Frehafer, 11/24/04)

Revision 5 - Incorporated PCR 03-3361 for CR 03-4025 to add design basis equipment used to maintain pressure and temperature in containment during a DBA. (Edgard Hernandez, 02/26/04)

Revision 4 – Incorporated PCR 03-0626 for PM 02-09-030 and PSL-ENG-SENS-02-044, R0 to provide protection of HVAC charcoal filters. (C.J. Wasik, 04/18/03)

Revision 3 – Changes made to reflect TS Amendment #127. (M. DiMarco, 09/20/02)

Revision 2 – Extended the allowed outage time for the containment vacuum relief lines from 4 hours to 72 hours for returning an inoperable containment vacuum relief line to operable status. (M. DiMarco, 06/06/02)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R. G. West Plant General Manager	08/30/01
Revision	**FRG Review Date**	**Approved By**	**Approval Date**
8		N/A Plant General Manager	
		Eric Katzman Authorized Approver	06/24/09
		N/A Authorized Approver (Minor Correction)	

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BASES FOR SECTION 3/4.6

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will ensure that the site boundary radiation doses are below the guidelines established for design basis accidents. /R8

In accordance with Generic Letter 91-08, "Removal of Component Component Lists from Technical Specifications," the opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a (41.8 psig) which results from the limiting design basis loss of coolant accident.

The surveillance testing for measuring leakage rates is performed in accordance with the Containment Leakage Rate Testing Program, and is consistent with the requirements of Appendix J of 10 CFR 50 Option B and Regulatory Guide 1.163 dated September, 1995, as modified by approved exemptions.

3/4.6 CONTAINMENT SYSTEMS (continued)

<u>**BASES**</u> (continued)

3/4.6.1 CONTAINMENT VESSEL (continued)

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.7 psi and (2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 43.4 psig. The limit of 0.4 psig for initial positive containment pressure will limit the total pressure to 43.99 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment temperature does not exceed the design temperature of 264°F during steam line break conditions and is consistent with the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

The limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 41.8 psig in the event of the limiting design basis loss of coolant accident. A visual inspection in accordance with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.1 CONTAINMENT VESSEL (continued)

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 48-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes devices to lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR 50.67 would not be exceeded in the vent of an accident during purging operations. /R8

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests. Leakage integrity testing does not apply to valves FCV-25-1 and FCV-25-6 because these valves provide shield building ventilation system integrity. FCV-25-1 and FCV-25-6 do not provide a containment isolation function and are not required by design to satisfy GDC-56 criteria for containment penetration isolation (see evaluation PSL-ENG-SENS-00-012).

3/4.6 CONTAINMENT SYSTEMS (continued)

<u>BASES</u> (continued)

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS

The OPERABILITY of the containment spray and cooling systems ensures that depressurization and cooling capability will be available to limit post-accident pressure and temperature in the containment to acceptable values. During a Design Basis Accident (DBA), at least one containment cooling train and one containment spray train are capable of maintaining the peak pressure and temperature within design limits. One containment spray train has the capability, in conjunction with the Iodine Removal System, to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analyses. To ensure that these conditions can be met considering single-failure criteria, two spray trains and two cooling trains must be OPERABLE.

The 72 hour action interval specified in ACTION 1.a and ACTION 1.d, and the 7 day action interval specified in ACTION 1.b take into account the redundant heat removal capability and the iodine removal capability of the remaining operable systems, and the low probability of a DBA occurring during this period. The 10 day constraint for ACTIONS 1.a and 1.b is based on coincident entry into two ACTION conditions (specified in ACTION 1.c) coupled with the low probability of an accident occurring during this time. If the system(s) cannot be restored to OPERABLE status within the specified completion time, alternate actions are designed to bring the unit to a mode for which the LCO does not apply. The extended interval (54 hours) specified in ACTION 1.a to be in MODE 4 includes 48 hours of additional time for restoration of the inoperable CS train, and takes into consideration the reduced driving force for a release of radioactive material from the RCS when in MODE 3. With two containment spray trains or any combination of three or more containment spray and containment cooling trains inoperable in MODES 1, 2, or Mode 3 with Pressurizer Pressure $\geq$ 1750 psia, the unit is in a condition outside the accident analyses and LCO 3.0.3 must be entered immediately. In MODE 3 with Pressurizer Pressure < 1750 psia, containment spray is not required.

The specifications and bases for LCO 3.6.2.1 are consistent with NUREG-1432, Revision 0 (9/28/92), Specification 3.6.6A (Containment Spray and Cooling Systems; Credit taken from iodine removal by the Containment Spray System), and the plant safety analyses.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS (continued)

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS (continued)

Ensuring that the containment spray pump discharge pressure is met satisfies the periodic surveillance requirement to detect gross degradation caused by impeller structural damage or other hydraulic component problems. Along with this requirement, Section XI of the ASME Code verifies the pump developed head at one point on the pump characteristic curve to verify both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. Surveillance Requirements are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the Iodine Removal System ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure a minimum of 50 ppm of N_2H_4 concentration available in the spray for a minimum of 6.5 hours per pump for a total of 13 hours to provide assumed iodine decontamination factors on the containment atmosphere during spray function and ensure a pH value of between 7.0 and 8.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 DELETED

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.4 DELETED

3/4.6.5 VACUUM RELIEF VALVES

BACKGROUND: The vacuum relief valves protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of the containment cooling system or the containment spray system. Multiple equipment failures or human errors are necessary to have inadvertent actuation.

The containment pressure vessel contains two 100% vacuum relief lines installed in parallel that protect the containment from excessive external loading. The vacuum relief lines are 24-inch penetrations that connect the shield building annulus to the containment. Each vacuum relief line is isolated by a pneumatically operated butterfly valve in series with a check valve located on the containment side of the penetration.

A separate pressure controller that senses the differential pressure between the containment and the annulus actuates each butterfly valve. Each butterfly valve is provided with an air accumulator that allows the valve to open following a loss of instrument air. The combined pressure drop at rated flow through either vacuum relief line will not exceed the containment pressure vessel design external pressure differential of 0.7 psid with any prevailing atmospheric pressure.

APPLICABLE SAFETY ANALYSES: Design of the vacuum relief lines involves calculating the effect of an inadvertent containment spray actuation that can reduce the atmospheric temperature (and hence pressure) inside containment. Conservative assumptions are used for all the pertinent parameters in the calculation The resulting containment pressure versus time is calculated, including the effect of the vacuum relief valves opening when their negative pressure setpoint is reached. It is also assumed that one vacuum relief line fails to open.

The containment was designed for an external pressure load equivalent to 0.7 psig. The inadvertent actuation of the containment spray system was analyzed to determine the resulting reduction in containment pressure. This resulted in a differential pressure between the inside containment and the annulus of 0.615 psid, which is less than the design load.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.5 VACUUM RELIEF VALVES (continued)

The vacuum relief valves must also perform the containment isolation function in a containment high-pressure event. For this reason, the system is designed to take the full containment positive design pressure and the containment design basis accident (DBA) environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO: The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of the containment spray system. Two vacuum relief lines are required to be OPERABLE to ensure that at least one is available, assuming one or both valves in the other line fail to open.

APPLICABILITY SAFETY ANALYSES: In MODES 1, 2, and 3 with pressurizer pressure equal to or greater than 1750 psia, the containment cooling features, such as the containment spray system, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur whenever these systems are OPERABLE due to inadvertent actuation of these systems. In MODES 1, 2, 3, and 4, the containment internal pressure is maintained between specified limits. Therefore, the vacuum relief lines are required to be OPERABLE in MODES 1, 2, 3, and 4 to mitigate the effects of inadvertent actuation of the containment spray system or containment cooling system.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. The containment spray system and containment cooling system are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief lines is not required in MODE 5 or 6.

ACTIONS: With one of the required vacuum relief lines inoperable, the inoperable line must be restored to OPERABLE status within 72 hours. The specified time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA. If the vacuum relief line cannot be restored to OPERABLE status within the required ACTION time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within the next 6 hours and to MODE 5 within the following 30 hours. The allowed ACTION times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.5 VACUUM RELIEF VALVES (continued)

SURVEILLANCE REQUIREMENTS: This SR references the Inservice Testing Program, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda and approved relief requests. Therefore, the Inservice Testing Program governs SR interval. The butterfly valve setpoint is 9.85±0.35 inches of water gauge differential.

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere and also reduces radioactive effluent releases to the environment during a fuel handling accident involving a recently irradiated fuel assembly in the spent fuel storage building. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 50.67 during LOCA conditions. /R8

The fuel handling accident analysis assumes a minimum post reactor shutdown decay time of 72 hours. Therefore, recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. This represents the applicability bases for fuel handling accidents. Containment closure will have administrative controls in place to assure that a single normal or contingency method to promptly close the primary or secondary containment penetrations will be available. These prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw the release from the postulated fuel handling accident in the proper direction such that it can be treated and monitored.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

With respect to Surveillance 4.6.6.1.b, this SR verifies that the required Shield Building Ventilation System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). /R8

3/4.6.6.2 SHIELD BUILDING INTEGRITY

SHIELD BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the shield building ventilation system, will ensure that the site boundary radiation doses are below the guidelines established for design basis. /R8

3/4.6.6.3 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide (1) protection for the steel vessel from the external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

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FPL

ST. LUCIE UNIT 2

TECHNICAL SPECIFICATIONS BASES ATTACHMENT 9 OF ADM-25.04

SAFETY RELATED

Section No. 3/4.7
Attachment No. 9
Current Revision No. 4
Effective Date 06/27/09

Title:

PLANT SYSTEMS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 4 – Incorporated PCR 08-6765 for CR 2007-32178 for Bases changes to Technical Specifications 155 for License Amendments 152 and 153. Procedure changes to implement AST were reviewed in ORG 07-041 on 5/29/07 as part of the license amendment submittal. (Author: Ken Frehafer)

AND

Incorporated PCR 08-6764 for CR 2008-23612 for Bases changes to Technical Specifications 155 for License Amendments 152 and 153. Procedure changes to implement Habitability were reviewed in ORG 07-041 on 5/29/07 as part of the license amendment submittal. (Author: Ken Frehafer)

Revision 3 - Incorporated PCR 06-1935 for PCM 05197 to update Unit 2 tech spec bases sections 3/4.4 and 3/4.7. (Modesto Jimenez, 06/28/06)

Revision 2 – Incorporated PCR 03-0626 for PM 02-09-030 and PSL-ENG-SENS-02-044, R0 to provide protection of HVAC charcoal filters. (C.J. Wasik, 04/18/03)

Revision 1 – Change per PSL-ENG-SENS-98-007, Rev. 0. (M. DiMarco, 09/23/02)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
Revision 4	FRG Review Date	Approved By N/A Plant General Manager	Approval Date
		Eric Katzman Authorized Approver	06/24/09
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S_2_OPS	
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BASES FOR SECTION 3/4.7

3/4.7 PLANT SYSTEMS

<u>BASES</u>

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1100 psia) of its design pressure of 1000 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and ASME Code for Pumps and Valves, Class II. The total relieving capacity for all valves on all of the steam lines is 12.49×10^6 lbs/hr which is 103.8% of the total secondary steam flow of 12.03×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip set-point reductions are derived on the following bases:

For two loop operation:

$$SP = \left[\frac{(X)-(Y)(V)}{X} \times (107.0) \right] - 0.9$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.1 SAFETY VALVES (continued)

107.0 = Power Level-High Trip Setpoint for two loop operation

0.9 = Equipment processing uncertainty

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.247×10^6 lbs/hr)

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.74×10^5lbs/hr)

Surveillance Requirement 4.7.1.1 verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The MSSV setpoints are 1000 psia +1/-3% (4 valves each header) and 1040 psia +1/-3% (4 valves each header) for OPERABILITY; however, the valves are reset to 1000 psia +/- 1% and 1040 psia +/- 1%, respectively, during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

The provisions for Specification 3.0.4 do not apply. This allows entry into and operation in MODE 3 prior to performing the Surveillance Requirement so that the MSSVs may be tested under hot conditions.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system pumps ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 320 gpm at a pressure of 1000 psia to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 500 gpm at a pressure of 1000 psia to the entrance of the steam generators. This capacity is sufficient to ensure adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the Unit 2 RCS at HOT STANDBY conditions for 4 hours followed by an orderly cooldown to the shutdown cooling entry temperature (350°F). The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The actual water requirements are 149,600 gallons for Unit 2 and 125,000 gallons for Unit 1. Included in the required volumes of water are the tank unusable volume of 9400 gallons and a conservative allowance for instrument error of 21,400 gallons.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will comply with the dose criterion provided in 10 CFR 50.67 in the event of a steam line rupture. The dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the safety analyses. /R4

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

The specified 6.75 second full closure time represents the addition of the maximum allowable instrument response time of 1.15 seconds and the maximum allowable valve stroke time of 5.6 seconds. These maximum allowable values should not be exceeded because they represent the design basis values for the plant.

3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.6 MAIN FEEDWATER LINE ISOLATION VALVES

The main feedwater line isolation valves are required to be OPERABLE to ensure that (1) feedwater is terminated to the affected steam generator following a steam line break and (2) auxiliary feedwater is delivered to the intact steam generator following a feedwater line break. If feedwater is not terminated to a steam generator with a broken main steam line, two serious effects may result: (1) the post-trip return to power due to plant cooldown will be greater with resultant higher fuel failure and (2) the steam released to containment will exceed the design.

When the main feedwater isolation valves (MFIVs) are closed or isolated, they are performing their required safety function, e.g., to isolate the main feedwater line. The 72 hour action completion time for one inoperable MFIV in one ore more main feedwater lines takes into account the redundancy afforded by the remaining operable MSIVs, and the low probability of an event occurring during this time period that would require isolation of the main feedwater flow paths. The 4 hour action completion time for two inoperable MFIVs in the same feedwater line is considered reasonable to close or isolate the affected flowpath. It is based on operating experience and the low probability of an event that would require main feedwater isolation during this time period.

The specified 5.15 second full closure time represents the addition of the maximum allowable instrument response time of 1.15 seconds and the maximum allowable valve stroke time of 4.0 seconds. These maximum allowable values should not be exceeded because they represent the design basis values for the plant.

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the atmospheric dump valves in the manual mode of operation is to ensure the atmospheric dump valves will be closed in the event of a steam line break. For the steam line break with atmospheric dump valve control failure event, the failure of the atmospheric dump valves to close would be a valid concern were the system to be in the automatic mode during power operations.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to 100°F and 200 psig are based on a steam generator RT_{NDT} of 20°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 INTAKE COOLING WATER SYSTEM

The OPERABILITY of the Intake Cooling Water System ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level is based on providing an adequate cooling water supply to safety-related equipment until cooling water can be supplied from Big Mud Creek.

Cooling capacity calculations are based on an ultimate heat sink temperature of 95°F. It has been demonstrated by a temperature survey conducted from March 1976 to May 1981 that the Atlantic Ocean has never risen higher than 86°F. Based on this conservatism, no ultimate heat sink temperature limitation is specified. (Note that with the implementation of the CCW heat exchanger performance monitoring program, the limiting ultimate heat sink temperature is treated as a variable with an upper limit of 95°F without compromising any margin of safety. System operation is maintained well within safety design limits for the service conditions of the heat exchanger.)

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The installation of the stoplogs ensures adequate protection for wave run-up effects where no permanent adjacent structures exist and provides protection to safety-related equipment. The maximum wave runup from the probable maximum flood (PMF) has been calculated to be elevation 18.0 feet Mean Low Water (MLW).

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the Control Room Emergency Air Cleanup System ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems total effective dose equivalent.

The control room envelope (CRE) is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The location of CREACS components and ducting within the CRE control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the CRE occupants, as demonstrated by occupant dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15.

/R4

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (continued)

In order for the CREACS to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CREACS must be OPERABLE to ensure that the CRE will remain habitable to limit operator exposure during and following a DBA.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem total effective dose equivalent - TEDE), or inadequate protection of CRE occupants from smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

/R4

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (continued)

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour allowable outage time (AOT) is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day AOT is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day AOT is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

In MODE 1, 2, 3, or 4, if the inoperable CREACS or the CRE boundary cannot be restored to OPERABLE status within the required AOT, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 30 hours. The AOT are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

When in MODES 5 and or 6, or during movement of irradiated fuel assemblies, with both CREACS trains inoperable or with one or more CREACS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

/R4

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (continued)

The Surveillance Requirement (SR) 4.7.7.e verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate in Modes 1, 2, 3, and 4, ACTION b must be taken. Required ACTION b.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F. These compensatory measures may also be used as mitigating actions as required by Required Action b.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY, as discussed in letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

3/4.7.8 ECCS AREA VENTILATION SYSTEM

The OPERABILITY of the ECCS Area Ventilation System ensures that cooling air is provided for ECCS equipment.

With respect to Surveillance 4.7.8.b, this SR verifies that the required ECCS Area Ventilation System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP).

/R4

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.9 SNUBBERS

All safety related snubbers are required to be OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip and 100 kip capacity manufactured by company "A" are of the same type. The same design mechanical snubber manufactured by company "B", for purposes of this Specification, would be of a different type, as would hydraulic snubbers for either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability, one of two sampling and acceptance criteria methods are used:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1.

3/4.7 CONTAINMENT SYSTEMS (continued)

<u>BASES</u> (continued)

3/4.7.9 SNUBBERS (continued)

Figures 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

All service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.11 DELETED

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 10 OF ADM-25.04 SAFETY RELATED	Section No. 3/4.8 Attachment No. 10 Current Revision No. 3 Effective Date 08/27/09

Title:

ELECTRICAL POWER SYSTEMS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 3 – Incorporated PCR 09-2643 to update EDG fuel oil testing ASTM standards. (Author: K.W. Frehafer)

Revision 2 – Implemented License Amendment 207 and 155. Procedure changes to implement EDG Fuel Oil Test Program LAR were reviewed in ORG 08-034 on 6/26/08 as part of the license amendment submittal. (Author: K.W. Frehafer)

Revision 1 – Implemented License Amendment 123. (K.W. Frehafer, 12/17/01)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

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3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional requirement to check that all required systems, subsystems, trains, components and devices (i.e., redundant features), that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. These redundant required features are those that are assumed to function to mitigate an accident, coincident with a loss of offsite power, in the safety analysis, such as the emergency core cooling system and auxiliary feedwater system. Upon discovery of a concurrent inoperability of required redundant features the feature supported by the inoperable EDG is declared inoperable. Thus plant operators will be directed to supported feature TS action requirements for appropriate remedial actions for the inoperable required features.

3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

The four hour completion time upon discovery that an opposite train required feature is inoperable is to provide assurance that a loss of offsite power, during the period that a EDG is inoperable, does not result in a complete loss of safety function of critical redundant required features. The four hour completion time allows the operator time to evaluate and repair any discovered inoperabilities. This completion time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The four hour completion time only begins on discovery that both an inoperable EDG exists and a required feature on the other train is inoperable.

TS 3.8.1.1, ACTION "b" provides an allowed outage/action completion time (AOT) of up to 14 days to restore a single inoperable diesel generator to operable status. This AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk-informed" AOT. Entry into this action requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the Administrative Procedure that implements the Maintenance Rule pursuant to 10 CFR 50.65.

All EDG inoperabilities must be investigated for common-cause failures regardless of how long the EDG inoperability persists. When one diesel generator is inoperable, required ACTIONS 3.8.1.1.b and 3.8.1.1.c provide an allowance to avoid unnecessary testing of EDGs. If it can be determined that the cause of the inoperable EDG does not exist on the remaining OPERABLE EDG, then SR 4.8.1.1.2.a.4 does not have to be performed. Eight (8) hours is reasonable to confirm that the OPERABLE EDG is not affected by the same problem as the inoperable EDG. If it cannot otherwise be determined that the cause of the initial inoperable EDG does not exist on the remaining EDG, then satisfactory performance of SR 4.8.1.1.2.a.4 suffices to provide assurance of continued OPERABILITY of that EDG. If the cause of the initial inoperability exists on the remaining OPERABLE EDG, that EDG would also be declared inoperable upon discovery, and ACTION 3.8.1.1.e would be entered. Once the failure is repaired (on either EDG), the common-cause failure no longer exists.

3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

4.8.1.1.2.c requires verification that the fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel Fuel Oil Program.

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057,

b. Verify in accordance with the tests specified in ASTM D975 that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89, or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥ 125°F, and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176 or a water and sediment content within limits when tested in accordance with ASTM D2709.

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975 are met for new fuel oil when tested in accordance with ASTM D975, except that the analysis for sulfur may be performed in accordance with ASTM D5453, ASTM D2622, or ASTM D3120. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs. /R3

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D6217 or ASTM D2276. This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. /R3

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

ASTM Standards: D4057; D975 and D975 Table 1; D1298; D4176; D2709; D2622; D6217; D5453; D3120; D2276. ASTM Standard "year" designations are located in Chemistry Procedures COP-05.10 and COP-07.16. /R3

This concludes the TS Bases discussion for SR 4.8.1.1.2.c.

3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9 “Selection of Diesel Generator Set Capacity for Standby Power Supplies,” March 10, 1971, and 1.108 “Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants,” Revision 1, August 1977, and 1.137, “Fuel Oil Systems for Standby Diesel Generators,” Revision 1, October 1979, Generic Letter 84-15, “Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,” dated July 2, 1984, and NRC staff positions reflected in Amendment No. 48 to Facility Operating License NPF-7 for North Anna Unit 2, dated April 25, 1985; as modified by Generic Letter 93-05, “Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,” dated September 27, 1993, and Generic Letter 94-01, “Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,” dated May 31, 1994.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, “Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants,” February 1978, and IEEE Std 450-1980, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.”

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

The OPERABILITY of the motor operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 11 OF ADM-25.04 SAFETY RELATED	Section No. 3/4.9 Attachment No. 11 Current Revision No. 4 Effective Date 06/25/04

Title:

REFUELING OPERATIONS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 4 – Incorporated PCR 04-1950 to delete BASES 3/4.9.7 and 3/4.9.12. (Glenn Adams, 06/22/04)

Revision 3 – Changes made to reflect TS Amendment #127. (M. DiMarco, 09/20/02)

Revision 2 – Changes made to reflect TS Amendment #122. (K.W. Frehafer, 11/30/01)

Revision 1 – Modified bases for Containment Building Penetrations in accordance with NRC SER "Containment Doors Open During Core Alterations" per approved License Amendment No. 120. (M. DiMarco, 11/08/01)

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

Revision	FRG Review Date	Approved By	Approval Date
0	08/30/01	R.G. West Plant General Manager	08/30/01
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BASES FOR SECTION 3/4.9

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value specified in the COLR for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value specified in the COLR includes a conservative uncertainty allowance of 50 ppm boron.

If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action. Suspension of CORE ALTERATIONS or positive reactivity additions shall not preclude moving a component to a safe position.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9 REFUELING OPERATIONS (continued)

BASES (continued)

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a recently irradiated fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. The fuel handling accident analysis assumes a minimum post reactor shutdown decay time of 72 hours. Therefore, recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. This represents the applicability bases for fuel handling accidents. Containment closure will have administrative controls in place to assure that a single normal or contingency method to promptly close the primary or secondary containment penetrations will be available. These prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw the release from the postulated fuel handling accident in the proper direction such that it can be treated and monitored.

FPL made the following regulatory commitment, which is consistent with NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 3, Section 11.3.6, *Assessment Methods for Shutdown Conditions*, subheading 11.3.6.5, *Containment – Primary (PWR)/Secondary (BWR).*

The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:

- During fuel handling/core alterations, ventilation system and radiation monitor ***availability*** (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases.

3/4.9 REFUELING OPERATIONS (continued)

BASES (continued)

3/4.9.4 CONTAINMENT PENETRATIONS (continued)

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of the "prompt methods" mentioned above are to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

Availability as defined by NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*, December 1991, relies on the definitions of **functional**, and **operable**. The NUMARC 91-06 definitions for these three terms follow.

- Available (Availability): 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 (Functionality): 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: The ability of a system to perform its specified function with all applicable TS requirements satisfied.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9 REFUELING OPERATIONS (continued)

BASES (continued)

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the refueling machine ensures that: (1) manipulator cranes will be used for movement of fuel assemblies, with or without CEAs, (2) each crane has sufficient load capacity to lift a fuel assembly, with or without CEAs, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 DELETED /R4

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operations.

3/4.9 REFUELING OPERATIONS (continued)

BASES (continued)

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION (continued)

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange with irradiated fuel in the core ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange with irradiated fuel in the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

The footnote providing for a minimum reactor coolant flow rate of ≥ 1850 gpm considers one of the two RCS injection points for a SDCS train to be isolated. The specified parameters include 50 gpm for flow measurement uncertainty, and 3°F uncertainty for RCS and CCW temperature measurements. The conditions of minimum shutdown time, maximum RCS temperature, and maximum temperature of CCW to the shutdown cooling heat exchanger are initial conditions specified to assure that a reduction in flow rate from 3000 gpm to 1800 gpm will not result in a temperature transient exceeding 140°F during conditions when the RCS water level is at an elevation ≥ 29.5 feet.

3/4.9.9 CONTAINMENT ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment isolation valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material resulting from a fuel handling accident of a recently irradiated fuel assembly from the containment atmosphere to the environment. Recently irradiated fuel is defined as fuel that has occupied parts of a critical reactor core within the previous 72 hours.

3/4.9 REFUELING OPERATIONS (continued)

BASES (continued)

3/4.9.4.10 and 3/4.9.11 WATER LEVEL – REACTOR VESSEL AND SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

The limit on soluble boron concentration in LCO 3/4.9.11 is consistent with the minimum boron concentration specified for the RWT, and assures an additional subcritical margin to the value of k_{eff} which is calculated in the spent fuel storage pool criticality safety analysis to satisfy the acceptance criteria of Specification 5.6.1. Inadvertent dilution of the spent fuel storage pool by the quantity of unborated water necessary to reduce the pool boron concentration to a value that would invalidate the criticality safety analysis is not considered to be a credible event. The surveillance frequency specified for verifying the boron concentration is consistent with NUREG-1432 and satisfies, in part, acceptance criteria established by the NRC staff for approval of criticality safety analysis methods that take credit for soluble boron in the pool water. The ACTIONS required for this LCO are designed to preclude an accident from happening or to mitigate the consequences of an accident in progress, and shall not preclude moving a fuel assembly to a safe position.

3/4.9.12 DELETED /R4

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 12 OF ADM-25.04 SAFETY RELATED	Section No. **3/4.10** Attachment No. **12** Current Revision No. **0** Effective Date **09/06/01**

Title:

SPECIAL TEST EXCEPTIONS

Responsible Department: **Licensing**

REVISION SUMMARY:

Revision 0 – Bases for Technical Specifications. (E. Weinkam, 08/30/01)

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0	08/30/01	R.G. West Plant General Manager	08/30/01	DATE DOCT PROCEDURE
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		Plant General Manager		COM COMPLETED ITM 0

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BASES FOR SECTION 3/4.10

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under reduced flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10 SPECIAL TEST EXCEPTIONS (continued)

BASES (continued)

3/4.10.5 CEA INSERTION DURING ITC, MTC, AND POWER COEFFICIENT MEASUREMENTS

This special test exception permits the CEA groups to be misaligned during such PHYSICS TESTS as those required to determine the (1) isothermal temperature coefficient, (2) moderator temperature coefficient, and (3) power coefficient.

FPL	ST. LUCIE UNIT 2 TECHNICAL SPECIFICATIONS BASES ATTACHMENT 13 OF ADM-25.04 SAFETY RELATED	Section No. 3/4.11 Attachment No. 13 Current Revision No. 0 Effective Date 09/06/01

Title:

RADIOACTIVE EFFLUENTS

Responsible Department: **Licensing**

REVISION SUMMARY:

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BASES FOR SECTION 3/4.11

3/4.11 RADIOACTIVE EFFLUENTS

BASES

Pages B 3/4 11-2 through B 3/4 11-3 (Amendment No. 61) have been deleted from the Technical Specifications. The next page is B 3/4 11-4.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."