

applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 200 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 28, 2003, describes certain future activities to be completed before the period of extended operation. FPL shall complete these activities no later than March 1, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on March 28, 2003, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed license. Until that update is complete, FPL may make changes to the programs described in such supplement without prior Commission approval, provided that FPL evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Sustained Core Uncovery Actions

Procedural guidance shall be in place to instruct operators to implement actions that are designed to mitigate a small-break loss-of-coolant accident prior to a calculated time of sustained core uncovery.

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DEFINITIONS

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

- a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or
- c. Reactor Coolant System leakage through a steam generator to the secondary system (Primary-to-secondary leakage).

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

1.16 The LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the cold leg temperature is $\leq 304^{\circ}\text{F}$ during heatup or $\leq 281^{\circ}\text{F}$ during cooldown and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 1.75 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

DEFINITIONS

OPERABLE – OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packing of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE – PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

REACTOR COOLANT SYSTEM

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the SG Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: *

- a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program;
 1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and
 2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.
- b. With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

- 4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.
- 4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

* Separate Action entry is allowed for each SG tube.

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REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following RCS leakage detection systems will be OPERABLE:

- a. The reactor cavity sump inlet flow monitoring system; and
- b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTION:

- a. With the required reactor cavity sump inlet flow monitoring system inoperable, perform a RCS water inventory balance at least once per 24* hours and restore the sump inlet flow monitoring system to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform a RCS water inventory balance at least once per 24* hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With all required monitors inoperable, enter LCO 3.0.3 immediately.
- d. The provisions of Specification 3.0.4 are not applicable if at least one of the required monitors is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The RCS leakage detection instruments shall be demonstrated OPERABLE by:

- a. Performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor at the frequencies specified in Table 4.3-3.
- b. Performance of the CHANNEL CALIBRATION of the required reactor cavity sump inlet flow monitoring system at least once per 18 months.

* Not required to be performed until 12 hours after establishment of steady state operation.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

- 3.4.6.2 Reactor Coolant System operational leakage shall be limited to:
- a. No PRESSURE BOUNDARY LEAKAGE,
 - b. 1 GPM UNIDENTIFIED LEAKAGE,
 - c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),
 - d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and
 - e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

- 4.4.6.2 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:
- a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

- b. Monitoring the containment sump inventory and discharge at least once per 12 hours,
- c. *Performance of a Reactor Coolant System water inventory balance at least once per 72 hours except when operating in the shutdown cooling mode,
- d. Monitoring the reactor head flange leakoff system at least once per 24 hours, and
- e. Verifying each Reactor Coolant System Pressure Isolation Valve leakage (Table 3.4.6-1) to be within limits:
 - 1. Prior to entering MODE 2 after refueling,
 - 2. Prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been performed in the previous 9 months,
 - 3. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.
 - 4. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.
- f. Whenever integrity of a pressure isolation valve listed in Table 3.4.6-1 cannot be demonstrated the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in each high pressure line having a leaking valve shall be recorded daily; and
- g. Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator at least once per 72 hours.**

* Not required to be performed until 12 hours after establishment of steady state operation. Not applicable to primary-to-secondary leakage.

** Not required to be performed until 12 hours after establishment of steady state operation.

ADMINISTRATIVE CONTROLS (continued)

k. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

1. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass $\leq 0.05\%$ when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

<u>ESF Ventilation System</u>	<u>Flowrate</u>
Control Room Emergency Ventilation	2000 \pm 200 cfm

2. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass $\leq 0.05\%$ when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

<u>ESF Ventilation System</u>	<u>Flowrate</u>
Control Room Emergency Ventilation	2000 \pm 200 cfm

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

<u>ESF Ventilation System</u>	<u>Penetration</u>	<u>RH</u>
Control Room Emergency Ventilation	$\leq 2.5\%$	70%

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

<u>ESF Ventilation System</u>	<u>Delta P</u>	<u>Flowrate</u>
Control Room Emergency Ventilation	< 4.15" W.G.	2000 \pm 200 cfm

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

i. Steam Generator (SG) Program

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

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

ADMINISTRATIVE CONTROLS (continued)

- I. Steam Generator (SG) Program (continued)
 - b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.
 1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
 2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total through all SGs and 0.5 gpm through any one SG.
 3. The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "Reactor Coolant System Operational Leakage."
 - c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.
 - d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

ADMINISTRATIVE CONTROLS (continued)

- i. Steam Generator (SG) Program (continued)
 - d. (continued)
 2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.
 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
 - e. Provisions for monitoring operational primary-to-secondary leakage.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

- 6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

- 6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment of the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.I, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG,
- b. Active degradation mechanisms found,
- c. Nondestructive examination techniques utilized for each degradation mechanism,
- d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
- e. Number of tubes plugged during the inspection outage for each active degradation mechanism,
- f. Total number and percentage of tubes plugged to date,
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and
- h. The effective plugging percentage for all plugging in each SG.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.