



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117  
License No. NPF-2

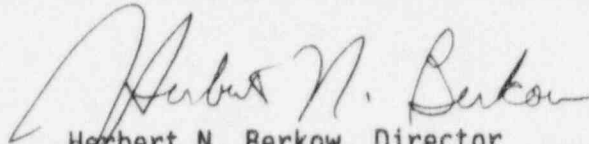
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated December 7, 1994, as supplemented May 31, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to the start of Unit 1, Cycle 14 operation.

FOR THE NUCLEAR REGULATORY COMMISSION



Herbert N. Berkow, Director  
Project Directorate II-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: September 28, 1995

ATTACHMENT TO LICENSE AMENDMENT NO.117

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages

3/4 4-10  
3/4 4-11  
3/4 4-12  
3/4 4-12a  
3/4 4-13  
3/4 4-23  
3/4 4-24  
3/4 4-25  
3/4 4-26  
B 3/4 4-3  
B 3/4 4-4  
B 3/4 4-5

Insert Pages

3/4 4-10  
3/4 4-11\*  
3/4 4-12  
3/4 4-12a  
3/4 4-13  
3/4 4-23  
3/4 4-24  
3/4 4-25  
3/4 4-26  
B 3/4 4-3  
B 3/4 4-4  
B 3/4 4-5

\* overflow page, no change

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.
  2. Tubes in those areas where experience has indicated potential problems.
  3. At least 3% of the total number of sleeved tubes in all three steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve.
  4. A tube inspection (pursuant to Specification 4.4.6.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
  5. Tube support plate indications left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during the following refueling outages.
- c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.
  2. The inspections include those portions of the tubes where imperfections were previously found.
- d. Implementation of the steam generator tube/tube support plate plugging criteria requires 100 percent bobbin coil inspection for hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

The results of each sample inspection shall be classified into one of the following three categories:

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.6.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

- a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.
- b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.6.3.a; the interval may then be extended to a maximum of once per 40 months.
- c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:
  1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.7.2.
  2. A seismic occurrence greater than the Operating Basis Earthquake.
  3. A loss-of-coolant accident requiring actuation of the engineered safeguards.
  4. A main steam line or feedwater line break.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.
2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.
3. Degraded Tube means a tube, including the sleeve if the tube has been repaired, that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.
4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.
5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective.
6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. For a tube that has been sleeved with a mechanical joint sleeve, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging. For a tube that has been sleeved with a welded joint sleeve, through wall penetration greater than or equal to 37% of sleeve nominal wall thickness in the sleeve between the weld joints requires the tube to be removed from service by plugging. This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 4.4.6.4.a.11 for the plugging limit applicable to these intersections.
7. Unserviceable describes the condition of a tube or sleeve if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.6.3.c, above.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube.
9. Tube Repair refers to mechanical sleeving, as described by Westinghouse report WCAP-11178, Rev. 1, or laser welded sleeving, as described by Westinghouse report WCAP-12672, which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure.
10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.
11. Tube Support Plate Plugging Limit is used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking confined within the thickness of the tube support plates. These criteria are applicable for the Fourteenth operating cycle only. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:
  - a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.
  - b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in 4.4.6.4.a.11.c below.
  - c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

- b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair of all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.6.5 Reports

- a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging or repair effort.
- b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:
  - 1. Number and extent of tubes and sleeves inspected.
  - 2. Location and percent of wall-thickness penetration for each indication of an imperfection.
  - 3. Identification of tubes plugged or repaired.
- c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a REPORTABLE EVENT and shall be reported pursuant to 10CFR50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.
- d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service (Mode 4) should any of the following conditions arise:
  - 1. If estimated leakage based on the actual end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous operating cycle.
  - 2. If circumferential crack-like indications are detected at the tube support plate intersections.
  - 3. If indications are identified that extend beyond the confines of the tube support plate.
  - 4. If the calculated conditional burst probability exceeds  $1.0 \times 10^{-2}$ , notify the NRC and provide an assessment of the safety significance of the occurrence.



REACTOR COOLANT SYSTEM

3/4.4.9 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.9 The specific activity of the primary coolant shall be limited to:

- a. Less than or equal to 0.5 microCurie per gram DOSE EQUIVALENT I-131;
- b. Less than or equal to  $100/\bar{E}$  microCurie per gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5

ACTION:

MODES 1, 2, and 3\*:

- a. With the specific activity of the primary coolant greater than 0.5 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with  $T_{avg}$  less than 500°F within 6 hours.
- b. With the specific activity of the primary coolant greater than  $100/\bar{E}$  microCurie per gram, be in at least HOT STANDBY with  $T_{avg}$  less than 500°F within 6 hours.

\* With  $T_{avg}$  greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

ACTION: (Continued)

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MODES 1, 2, 3, 4, and 5:

- a. With the specific activity of the primary coolant greater than 0.5 microCurie per gram DOSE EQUIVALENT I-131 or greater than  $100/\bar{E}$  microCuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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4.4.9 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4  
PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE  
AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS FREQUENCY</u>	<u>MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED</u>
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 0.5 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 or $100/\bar{E}$ $\mu\text{Ci}/\text{gram}$ , and  b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1#, 2#, 3#, 4#, 5#  1, 2, 3

# Until the specific activity of the primary coolant system is restored within its limits.

\* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

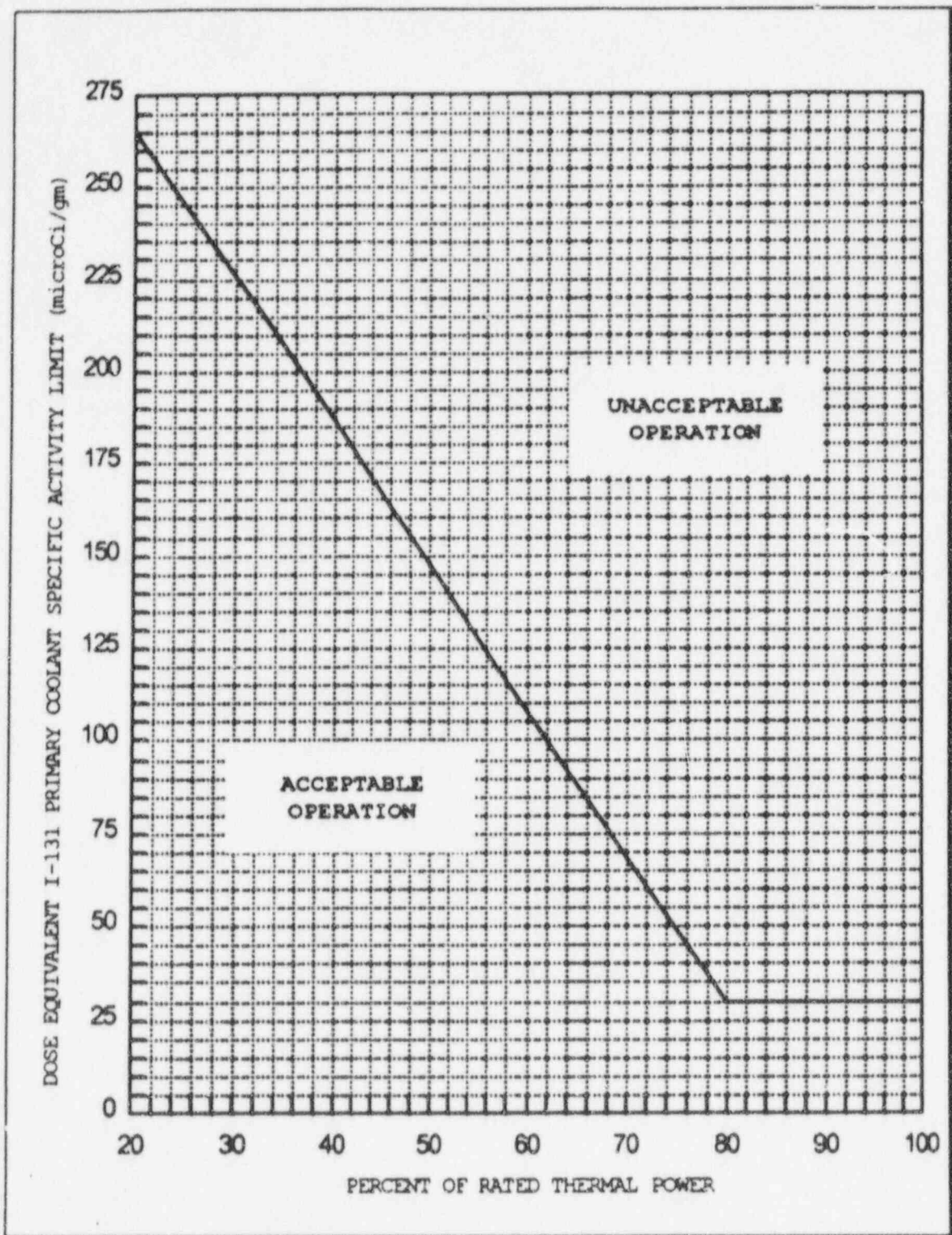


FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.5  $\mu$ Ci/gram Dose Equivalent I-131

REACTOR COOLANT SYSTEM  
BASES

3/4.4.6 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 140 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operational leakage of this magnitude can be readily detected by existing Farley Unit 1 radiation monitors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

The repair limit for ODSCC at tube support plate intersections is based on the analysis contained in WCAP-12871, Revision 2, "J. M. Farley Units 1 and 2 SG Tube Plugging Criteria for ODSCC at Tube Support Plates," and documentation contained in EPRI Report TR-100407, Revision 1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." The application of this criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable Part 100 limits are not exceeded.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. If a sleeved tube is found to have through wall penetration of greater than or equal to 31% for the mechanical sleeve and 37% for the laser welded sleeve of sleeve nominal wall thickness in the sleeve, it must be plugged. The 31% and 37% limits are derived from R.G. 1.121 calculations with 20% added for conservatism. The portion of the tube and the sleeve for which indications of wall degradation must be evaluated can be summarized as follows:

## REACTOR COOLANT SYSTEM

### BASES

#### 3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

##### 3/4.4.7.1 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.

##### 3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation valves is IDENTIFIED LEAKAGE and will be considered a portion of the allowed limit.

The total steam generator tube leakage limit of 420 gallons per day for all steam generators and 140 gallons per day for any one steam generator ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The limits are consistent with the assumptions used in the analysis of these accidents. The 140 gallons per day leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.8 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.

#### 3/4.4.9 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 an appropriately small fraction of Part 100 limits in the event of primary-to-secondary leakage as a result of a steamline break.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.5 microCuries/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.