



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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94  
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated May 28, 1993, as supplemented on July 29, 1993, September 14, 1993 and September 22, 1993 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-8 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 94 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in dark ink, appearing to read 'Singh Bajwa' with a stylized flourish at the end.

Singh Bajwa, Acting Director  
Project Directorate II-1  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical  
Specifications

Date of Issuance: October 20, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

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 4-12  
3/4 4-12a  
3/4 4-17  
3/4 4-17a  
3/4 4-23  
3/4 4-24  
3/4 4-26  
B3/4 4-3  
B3/4 4-3a  
B3/4 4-3b  
B3/4 4-5

Insert Pages

3/4 4-12  
3/4 4-12a  
3/4 4-17  
3/4 4-17a  
3/4 4-23  
3/4 4-24  
3/4 4-26  
B3/4 4-3  
B3/4 4-3a  
B3/4 4-3b  
B3/4 4-5

## 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. This definition does not apply to the area of the tubesheet region below the F\* distance in the F\* tubes. 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. At tube support plate intersections, the repair limit for the Tenth Operating Cycle is based on maintaining steam generator tube serviceability as described below:
  - a. An eddy current examination using a bobbin probe of 100% of the hot and cold leg steam generator tube support plate intersections will be performed for tubes in service.
  - b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to 1.0 volt will be allowed to remain in service.

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS (Continued)

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- c. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 1.0 volt will be repaired or plugged except as noted in 4.4.6.4.a.6.d below.
  - d. 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 1.0 volt but less than or equal to 3.6 volts may remain in service if a rotating pancake coil probe (RPC) inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than 3.6 volts will be plugged or repaired.
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.
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.

## REACTOR COOLANT SYSTEM

### OPERATIONAL LEAKAGE

#### LIMITING CONDITION FOR OPERATION

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3.4.7.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 GPM UNIDENTIFIED LEAKAGE,
- c. For the Tenth Operating Cycle only, primary-to-secondary leakage through all steam generators shall be limited to 450 gallons per day and 150 gallons per day through any one steam generator.

For subsequent cycles, 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

- d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and
- e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of  $2235 \pm 20$  psig.
- f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of  $2235 \pm 20$  psig.

APPLICABILITY: MODES 1, 2, 3 and 4

#### ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 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 specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS

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4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

- a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.
- b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

## REACTOR COOLANT SYSTEM

### 3/4.4.9 SPECIFIC ACTIVITY

#### LIMITING CONDITION FOR OPERATION

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

- a. Less than or equal to 0.25 microCurie per gram DOSE EQUIVALENT I-131 for the Tenth Operating Cycle only;
- b. Less than or equal to 1.0 microCurie per gram DOSE EQUIVALENT I-131 for subsequent cycles;
- c. 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. For the Tenth Operating Cycle only, with the specific activity of the primary coolant greater than 0.25 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. For subsequent cycles, with the specific activity of the primary coolant greater than 1.0 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.
- c. 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.

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



## REACTOR COOLANT SYSTEM

ACTION: (Continued)

MODES 1, 2, 3, 4, AND 5

- a. For the Tenth Operating Cycle only, with the specific activity of the primary coolant greater than 0.25 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/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.
- b. For subsequent cycles, with the specific activity of the primary coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/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.

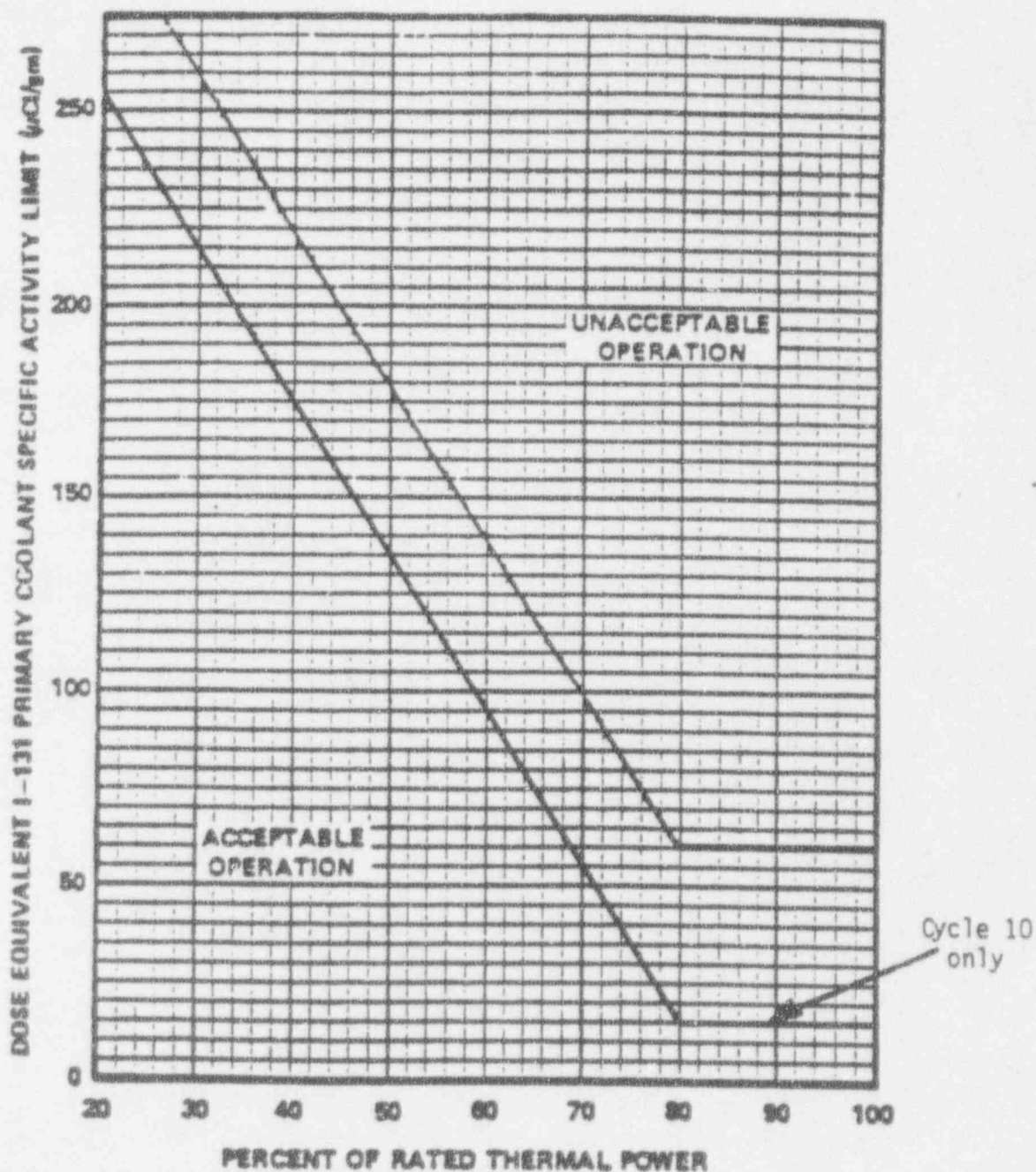


FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity  $> 1.0 \mu\text{Ci/gram}$  Dose Equivalent I-131

(Activity  $> .25 \mu\text{Ci/gram}$  Dose Equivalent I-131 for Cycle 10 only.)

## 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 = 500 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. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. 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.

For the Tenth Operating Cycle only, the repair limit for tubes with flaw indications contained within the bounds of a tube support plate has been provided to the NRC in Southern Nuclear Operating Company letter dated July 29, 1993. The repair limit 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." The application of this criteria is based on limiting primary-to-secondary leakage during a steam line break to less than 1 gallon per minute. Primary-to-secondary leakage during this cycle only is limited to 150 gallons per day per steam generator during normal operation.

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

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#### a. Mechanical

1. Indications of degradation in the entire length of the sleeve must be evaluated against the sleeve plugging limit.
2. Indication of tube degradation of any type including a complete guillotine break in the tube between the bottom of the upper joint and the top of the lower roll expansion does not require that the tube be removed from service.
3. The tube plugging limit continues to apply to the portion of the tube in the entire upper joint region and in the lower roll expansion. As noted above, the sleeve plugging limit applies to these areas also.
4. The tube plugging limit continues to apply to that portion of the tube above the top of the upper joint.

#### b. Laser Welded

1. Indications of degradation in the length of the sleeve between the weld joints must be evaluated against the sleeve plugging limit.
2. Indication of tube degradation of any type including a complete break in the tube between the upper weld joint and the lower weld joint does not require that the tube be removed from service.
3. At the weld joint, degradation must be evaluated in both the sleeve and tube.
4. In a joint with more than one weld, the weld closest to the end of the sleeve represents the joint to be inspected and the limit of the sleeve inspection.
5. The tube plugging limit continues to apply to the portion of the tube above the upper weld joint and below the lower weld joint.

F\* tubes do not have to be plugged or repaired provided the remainder of the tube within the tubesheet that is above the F\* distance is not degraded. The F\* distance is equal to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation. Included in this distance is an allowance of 0.25 inch for eddy current elevation measurement uncertainty.

Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

## REACTOR COOLANT SYSTEM

### BASES

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Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to 10 CFR 50.73 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision to the Technical specifications, if necessary.

## REACTOR COOLANT SYSTEM

### BASES

#### 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 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 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. 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 Farley site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

For the Tenth Operating Cycle only, the limitations on the specific activity of the primary coolant have been reduced. The reduction in specific activity limits continue to 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 steam line break.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 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.