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December 20, 1979

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Docket No. 50-281

Mr. W. L. Proffitt
Senior Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. DPR-37 for Unit No. 2 of the Surry Power Station. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 30, 1976 as supplemented May 24, 1979.

The amendment incorporates steam generator inservice inspection requirements and reactor coolant and secondary coolant iodine radioactivity concentration limits into the Technical Specifications and deletes a license condition that previously governed steam generator inspections.

These changes have been made to provide inservice inspection requirements suitable for essentially new steam generators in accordance with Regulatory Guide 1.83. Previous requirements were tailored to deteriorating steam generator tubes which were removed as a part of the steam generator repair program approved on January 19, 1979.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

(Signed)

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
Amendment No. 54 to DPR-37
Safety Evaluation
Notice of Issuance

cc: w/enclosures
See next page

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

December 20, 1979

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Sincerely,

A. Schwencer

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Division of Operating Reactors

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See next page

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated December 30, 1976 as supplemented May 24, 1979, 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 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.

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2. Accordingly, the license is amended by deleting paragraph E Steam Generator Inspection and by changes to the Technical Specifications as indicated in the attachment to the license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 54, are hereby incorporated in the license. The licensee 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. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: December 20, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
ii	ii
	3.1-13a
3.1-15	3.1-15
	3.1-15a
3.1-17	3.1-17
	3.1-17a
3.6-2	3.6-2
3.6-5	3.6-5
4.1-10	4.1-10
	4.1-10a
	4.19-1
	4.19-2
	4.19-3
	4.19-4
	4.19-5
	4.19-6
	4.19-7
	4.19-8
	4.19-9
	4.19-10
	4.19-11
	4.19-12

Appendix A-1

6. If the primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System exceeds 1 gpm total and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System, reduce the leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor.

References

FSAR, Section 4.2.7 - Reactor Coolant System Leakage

FSAR, Section 14.3.2 - Rupture of a Main Steam Pipe

D. Maximum Reactor Coolant Activity

Specifications

1. The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed $100/\bar{\bar{E}}$ μCi/cc whenever the reactor is critical or the average temperature is greater than 500°F, where $\bar{\bar{E}}$ is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection.

2. The specific activity of the reactor coolant shall be limited to $\leq$ 1.0 μCi/cc DOSE EQUIVALENT I-131 whenever the reactor is critical or the average temperature is greater than 500°F.

3. The requirements of D-2 above may be modified to allow the specific activity of the reactor coolant >1.0 μCi/cc DOSE EQUIVALENT I-131 but less than 10.0 μCi/cc DOSE EQUIVALENT I-131, ~~operation may continue~~ for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. With the specific activity of the reactor coolant >1.0 μCi/cc DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10.0 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.

4. If the specific activity of the reactor coolant exceeds 1.0 μCi/cc DOSE EQUIVALENT I-131 or $100/\bar{E}$ μ/Ci/cc, a report shall be prepared and submitted to the Commission pursuant to Specification 6.6.2.b(2). This report shall contain the results of the specific activity analysis together with the following information:

   a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

   b. Fuel burnup by core region,

   c. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

   d. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

   e. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/cc DOSE EQUIVALENT I-131.

boundary would be 0.30 Rem whole body and 0.28 Rem thyroid. Thus, these doses are well below the guidelines suggested in 10CFR100.

Permitting reactor operation to continue for limited time periods with the reactor coolant's specific activity >1.0 µCi/cc but < 10.0 µCi/cc DOSE EQUIVALENT I-131 accomodates possible iodine spiking phenomenon which may occur following changes in thermal power. Operation within these limits must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed may slightly increase the 2 hour thyroid dose at the site boundary following a postualted steam generator tube rupture. The basis for the 500°F temperature contained in the Specification is that the saturation pressure corresponding to 500°F, 680.8 psia, is well below the pressure at which the atmospheric relief valves on the secondary side could be actuated.

Measurement of $\bar{E}$ will be performed at least twice annually. Calculations required to determine $\bar{E}$ will consist of the following:

1. $\bar{E}$ shall be the average (weighed in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes.

3. A calculation of $\bar{E}$ by appropriate weighing of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/cc) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

E. Minimum Temperature for Criticality

Specifications

2. A minimum of 96,000 gal of water shall be available in the tornado missle protected condensate storage tank to supply emergency water to the auxiliary feedwater pump suctions.
3. All main steam line code safety valves, associated with steam generators in unisolated reactor coolant loops, shall be operable.
4. System piping and valves required for the operation of the components enumerated in Specification B.1, 2, and 3 shall be operable.

C. The iodine - 131 activity in the secondary side of any steam generator, in an unisolated reactor coolant loop, shall not exceed 9 curies. Also the specific activity of the secondary coolant system shall be ≤ 0.10 μCi/cc DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection and in the Cold Shutdown Condition within the following 30 hours.

D. The requirements of Specification B-2 above may be modified to allow utilization of protected condensate storage tank water with the auxiliary steam generator feed pumps provided the water level is maintained above 60,000 gallons, sufficient replenishment water is available in the 300,000 gallon condensate storage tank, and replenishment of the protected condensate storage tank is commenced within two hours after the cessation of protected condensate storage tank water consumption.

BASIS

A reactor which has been shutdown from power requires removal of core residual heat. While reactor coolant temperature or pressure is greater than 350°F or

The steam generator's specific iodine - 131 activity limit is calculated by dividing the total activity limit of 9 curies by the water volume of a steam generator. At full power, with a steam generator water volume of 47.6 M^3, the specific iodine - 131 limit would be .18 μCi/cc; at zero power, with a steam generator water volume of 101 M^3, the specific iodine - 131 limit would be .089 μCi/cc

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10CFR Part 100 limits in the event of a steam line rupture.

References

FSAR Section 4	Reactor Coolant System
FSAR Section 9.3	Residual Heat Removal System
FSAR Section 10.3.1	Main Steam System
FSAR Section 10.3.2	Auxiliary Steam System
FSAR Section 10.3.5	Auxiliary Feedwater Pumps
FSAR Section 10.3.8	Vent and Drain Systems
FSAR Section 14.3.2.5	Environmental Effects of a Steam Line Break

TABLE 4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1. Reactor Coolant Liquid Samples	Radio-chemical Analysis (1)	Monthly (5)	
	Gross Activity (2)	5 days/week (5)	9.1
	Tritium Activity	Weekly (5)	9.1
	*Chemistry (Cl, F & O_2)	5 days/week	4
	*Boron Concentration	Twice/week	9.1
	$\bar{E}$ Determination	Semiannually (3)	
	DOSE EQUIVALENT I-131	Once/2 weeks (5)	
	Radio-iodine Analysis (including I-131, I-133 & I-135)	Once/4 hours (6) and (7) below	
2. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly	6
3. Boric Acid Tanks	*Boron Concentration	Twice/week	9.1
4. Boron Injection Tank	Boron Concentration	Twice/week	6
5. Chemical Additive Tank	NaOH Concentration	Monthly	6
6. Spent Fuel Pit	*Boron Concentration	Monthly	9.5
7. Secondary Coolant	Fifteen minute degassed β and γ activity (4)	Once/72 hours	10.3
	DOSE EQUIVALENT I-131	Monthly (4) Semiannually (8)	
8. Stack Gas Iodine and Particulate Samples	*I-131 and particulate radioactive releases	Weekly	
9. Accumulator	Boron Concentration	Monthly	6.2

*See Specification 4.1.D

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) A gross beta-gamma degassed activity analysis shall consist of the quantitative measurement of the toal radioactivity of the primary coolant in units of μCi/cc.

(3) $\bar{E}$ determination will be started when the gross gamma degassed activity of radionuclides with half-lives greater than 15 minutes analysis indicates $\geq$ 10μCi/cc. Routine sample(s) for $\bar{E}$ analyses shall only 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.

(4) If the fifteen minute degassed beta and gamma activity is 10% or more of the 9 Curie limit given in Specification 3.6.C, a DOSE EQUIVALENT I-131 analysis will be performed.

(5) When reactor is critical and average primary coolant temperature $\geq$350°F.

(6) Whenever the specific activity exceeds 1.0 μCi/cc DOSE EQUIVALENT I-131 or $100/\bar{E}$ μCi/cc and until the specific activity of the reactor coolant system is restored within its limits.

(7) One sample between 2 & 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one hour period provided the average primary coolant temperature $\geq$350°F.

(8) When the fifteen minute degassed beta and gamma activity is less than 10% of the 9 Curie limit given in Specification 3.6.C.

4.19 STEAM GENERATOR INSERVICE INSPECTION

Applicability

Applies to the periodic inservice inspection of the steam generators.

Objective

To provide assurance of the continued integrity of the steam generator pressure boundaries.

Specifications

A. Each steam generator shall be demonstrated operable pursuant to Specification 3.1.A.2 by performance of the following augmented inservice inspection program and the requirement of Specification 4.2.A.

B. Steam Generator Sample Selection and Inspection - Each steam generator shall be determined operable during shutdown by selecting and inspection at least the minimum number of steam generators specified in Table 4.19-1.

C. Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in

Table 4.19-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.19.D and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.19.E. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations > 20%.

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.19.E.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 inspection, this shall be recorded and an

adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.19-2) 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.

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

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 must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

D. Inspection Frequencies - The above 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 4.19-2 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 4.19.D.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 4.19-2 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 3.1.C.6.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A major main steam line or feedwater line break.

E. Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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.

3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused be degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect is 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 4.19.D.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.

9. 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 using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined operable after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.19-2.

F. Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which the inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specificatin 6.6 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

BASIS

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 unit is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes.If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitatin of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage of 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operatin will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plant 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.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.19.E.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.6 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 of the Technical Specifications, if necessary.

TABLE 4.19-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes			
No. of Steam Generators	Two	Three		Two	Three		
First Inservice Inspection	All			One	Two		
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]		

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

TABLE 4.19-2

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specificatin 6.6.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.6.	N/A	N/A

$S = 3 \frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 54 TO

FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 2

DOCKET NO. 50-281

Introduction

By letter dated December 30, 1976, as supplemented May 24, 1979, Virginia Electric and Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating License Nos. DPR-32 and DPR-37 for Surry Power Station, Unit Nos. 1 and 2.

Discussion

In August 1974, we requested that the licensee submit proposed Technical Specification changes that would establish requirements for a program of steam generator tube inspection. To provide guidance in developing an inspection program at that time, the licensee was to refer to Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," dated June 1974 (R.G. 1.83). The licensee submitted a program for Surry on November 18, 1974. However, we made a decision to delay requiring Technical Specification incorporation of the program at that time because of a need to revise R.G. 1.83 to reflect developments in the state-of-the-art of steam generator tube inspection techniques and to more directly take into account the inspection experience that was being gained at operating plants. In making that decision we took into account the industry wide practice which already included voluntary inspection of steam generator tubes that in many respects, was comparable to inspections that R.G. 1.83 specified. Revision 1 to R.G. 1.83 was issued after receiving comments from the industry. We are now taking steps to require incorporation of steam generator tube inservice inspections into the Technical Specifications. By letter dated December 30, 1976 as supplemented May 24, 1979, the licensee proposed Technical Specifications which reflect the provisions of R.G. 1.83, Revision 1, with exceptions as discussed with the NRC staff.

The Technical Specifications proposed for the Surry steam generator tube inspections are, therefore, in general agreement with R.G. 1.83, Rev. 1, dated July 1975, but deviate in those areas where we have determined that the overall inspection program would be enhanced over that covered in R.G. 1.83, Rev. 1.

In addition to the proposed changes to the Technical Specifications to implement steam generator inservice inspection, the activity requirements of Appendix A-1 to the Technical Specification are being incorporated into the text of the Technical Specifications with some changes as discussed with the licensee.

The proposed changes to the Technical Specifications discussed in this Safety Evaluation Report are intended to be applicable to both Surry 1 and Surry 2. However, these changes will not be applied to Surry 1 until startup after steam generator repair on Surry 1. In the interim, inspection requirements on that facility are governed by existing license conditions.

Evaluation

1. Surveillance Requirements for Steam Generator Tubes

Structures, systems, and components, important to safety of a nuclear power plant are designed, fabricated, constructed, and tested so as to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. To continuously maintain such assurance, General Design Criterion 32 requires that components which are part of the reactor coolant pressure boundary (RCPB) be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The steam generator tubing is part of the RCPB and is an important part of a major barrier against fission product release to the environment. It also acts as a barrier against steam release to the containment in the event of a Loss of Coolant Accident (LOCA). For this reason, a program of periodic inservice inspection is being established to assure the continued integrity of the steam generator tubes over the service life of the plant.

Generally, the major elements of the steam generator tube inservice inspection program for Surry consist of specified: (a) sample selection, (b) examination methods, (c) inspection intervals, (d) acceptance criteria, and (e) reporting requirements. Each of these major elements of the program is separately evaluated below.

(a) Sample Selection

The proposed sampling program is patterned after R.G. 1.83, Rev. 1, except for those deviations that we have determined will improve the program and thereby reduce the potential radiation exposures especially for personnel that must perform the inspections. The sampling procedures are contained in

Table 4.19-2 of the proposed Technical Specifications. The principal deviations from R.G. 1.83, Rev. 1, supplementary sampling requirements are evaluated below.

(i) Regulatory Position C.5.a, "Supplementary Sampling Requirements," recommends that if the eddy current inspection results during an inservice inspection indicate any tubes with previously undetected imperfections of 20% or greater depth, additional steam generators, if any, should be inspected. In other words, because of a single tube in one steam generator with a previously undetected imperfection of 20%, or greater, depth, but still well below the plugging limit, all steam generators in the unit would be inspected. This would be unreasonably severe and would increase unnecessarily the radiation exposures of inspection personnel. The supplementary sampling requirements, as modified, would still require inspection of additional steam generators where significant, but only if the inspection results of the particular steam generator fall in category C-3 which is defined in Specification 4.19.C as "more than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective." By thus minimizing the inspection of other steam generators, the exposure to personnel can be kept low as is reasonably achievable.

(ii) Revision 1 of R.G. 1.83 recommends additional tube inspections in a steam generator if the inspection of a sample of tubes results in more than 10% of the tubes in that initial sample having detectable wall penetration of greater than 20% or if one or more tubes in the sample have an indication in excess of the plugging limit. The first additional inspection recommended by the guide would be the inspection of an additional 3% of the tubes in that steam generator concentrating on those areas where imperfections have been found. If 10% of these additionally inspected tubes fail to meet the criteria applied to the initial inspection sample, a third sample consisting of at least 6% of the tubes in that steam generator in the area of the imperfections would be expected.

For additional inspections, if required after the initial inspections, the program set forth in the Surry proposed Technical Specifications would require that all tubes in the affected steam generator be inspected and, further, that twice the number of tubes initially inspected in the affected steam generator be inspected in each of the other steam generators. Again, if more than 10% of the tubes inspected in any steam generator have indications of wall thinning of greater than 20% or more than 1% of the tubes are defective a third inspection is required of all tubes in each steam generator with such indications. The primary purpose of the additional inspections is to confirm the initial inspection results and to ensure steam generator integrity. By requiring that all tubes in the affected steam generator be inspected, and by requiring that the other steam generators be inspected when problems are detected, the Surry program represents an improvement to R.G. 1.83, Rev. 1.

Based on the considerations discussed above, we have concluded that the sample selection scheme proposed by the licensee is acceptable.

(b) Examination Method

The proposed examination methods, as modified by the NRC staff and concurred in by the licensee, include nondestructive examination by eddy current testing. The specified methods are capable of locating and identifying stress corrosion cracks and tube wall thinning from chemical wastage, mechanical damage or other causes. Based on our review of these methods, and experience gained using these methods by the industry, we have concluded that the examination methods are acceptable.

(c) Inspection Intervals

The proposed inspection intervals are compatible with those recommended in R.G. 1.83, Rev. 1; and thus, are acceptable.

(d) Acceptance Criteria

The principal parameter used to determine whether any one steam generator tube is acceptable for continued service is the measured imperfection depth. A tube plugging limit has been established and defined in the Technical Specifications as the imperfection depth beyond which the tube must be removed from service. The plugging limit is 40% degradation of the nominal tube wall-thickness.

The plugging limit is based on (1) the minimum tube wall thickness needed to maintain steam generator tube integrity during the limiting stress loadings associated with a LOCA combined with a Safe Shutdown Earthquake (SSE), and (2) an operational allowance to account for the time interval between inspections. Based on other evaluations made by the NRC staff[1], analyses performed by Westinghouse on steam generator tube designs similar to the Surry tube design, and Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator tubes, we conclude that the proposed 40% degradation plugging limit will ensure the required factors of safety of three against tube rupture under normal operating conditions and will provide a margin of safety consistent with Section III of the ASME Boiler and Pressure Vessel Code under postulated accident conditions including LOCA & SSE. Furthermore, the proposed plugging limit includes a sufficient thickness degradation allowance to compensate for possible continued degradation between inservice inspections. Therefore, we find that the proposed 40% degradation plugging limit is acceptable.

(e) Reporting of Inspection Results

Regulatory Position C.7.d of R.G. 1.83, states that a licensee should report to the Commission, for resolution and approval, proposed remedial action if the inspection results excees the limits specified in the Guide. It also states that additional sampling and more frequent inspection may be required. The proposed Technical Specifications, as modified by the NRC staff and concurred in by the licensee, clearly specify additional inspections the licensee must perform for those inspection results that fall in Categories C-1 and C-2. Immediate reporting of these results would not be required. Immediate reporting, as indicated by Table TS 4.19-2 would be required only if the inspection results reached the relatively greater level of tube degradation and defect found in Category C-3.

We conclude that the above described reporting requirements, as proposed by the licensee and modified by us, are reasonable and will facilitate reporting of pertinent information without unnecessarily increasing plant downtime, and thus constitute an acceptable alternative method for complying with the Commission's regulations.

In summary, we have concluded that the proposed steam generator tube inservice inspection program will provide added assurance of the continued integrity of the steam generator tubes, and thus is acceptable.

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[1] Supplemental Testimony of James P. Knight before the Atomic Safety and Licensing Appeal Board In The Matter of Northern States Power Company, Docket Nos. 50-282/306.

2. Reactor Coolant to Steam Generator Secondary Side Leak Rate Limit

(a) The existing license condition of 0.3 gpm specifies a reactor coolant leak rate limit. The proposed change would specify a reactor coolant leakage rate limit of 500 GPD (about 0.35 GPM) from any one steam generator, and would require that the leakage be reduced to within limits within 4 hours or that the plant be placed in hot shutdown within the next 6 hours and in cold shutdown within 30 hours if the leak rate is exceeded.

The limit of 0.3 gpm was established to assure an early shutdown for inspection upon detection of very small leaks taking into account the state of degradation of the plants steam generators prior to shutdown to accomplish the steam generator repair program in which the entire lower assemblies of the steam generators, including all tubes, were replaced.

These new lower assemblies are now in place in Unit 2 and constitute essentially new condition steam generators insofar as the primary-to-secondary pressure boundary is concerned. The standard NRC Technical Specifications for Westinghouse plants all use this 500 GPD limit for new steam generators.

We have, therefore, concluded that a 500 GPD leakage rate limit for each Surry 2 steam generator is acceptable at this time.

The proposal to move the steam generator inservice inspection requirement from the body of the license to the Appendix A Technical Specifications should be accompanied by a similar relocation of the related reactor coolant and secondary coolant activity limits incorporated into the Technical Specifications. These limits should be essentially the same as they are now with the exception of changing the total specific activity limit of the reactor coolant to $100/\bar{E}$ from $41/\bar{E}$ for nuclides with half-lives of more than 15 minutes. These changes are necessary to be consistent with the specifications used for other cases and with the standard NRC Technical Specifications applied to other Westinghouse reactors. The licensee agrees with these changes.

These limitations on reactor coolant specific activity will ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steady state primary-to-secondary steam generator reactor coolant leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a conservative application as a parametric evaluation by the NRC of typical sites. The NRC is finalizing site specific criteria which will be used as the basis for this reevaluation of the specific activity limits of the Surry site. This reevaluation may result in higher limits for the Surry site.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 20, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 54 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company, which revised Technical Specifications for operation of the Surry Power Station, Unit No. 2 (the facility) located in Surry County, Virginia. The amendment is effective as of the date of issuance.

The amendment incorporates steam generator inservice inspection requirements and reactor coolant and secondary coolant iodine radioactive concentration limits into the Technical Specifications and deletes a license condition that previously governed steam generator inspections.

These changes have been made to provide inservice inspection requirements suitable for essentially new steam generators in accordance with Regulatory Guide 1.83. Previous requirements were tailored to deteriorating steam generator tubes which were removed as a part of the steam generator repair program approved on January 19, 1979.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendment dated December 30, 1976 as supplemented May 24, 1979, (2) Amendment No. 54 to License No. DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 20th day of December, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors