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REID,R.W. Operating Reactors Branch 4

SUBJECT: Forwards proposed Tech Spec revision for review approval. Objective of proposed revision is to define inservice surveillance program for steam generator tubing.

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

January 23, 1979

TELEPHONE: AREA 704
373-4083

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. W. Reid, Chief
Operating Reactors Branch #4

Re: Oconee Nuclear Station
Docket Nos. 50-269, 270, 287

Dear Mr. Denton:

In response to an NRC request of September 21, 1976, Duke Power Company submitted a proposed Technical Specification revision on November 30, 1976, covering inservice inspection of steam generators. This proposal was modeled on the ASME Section XI Winter 1975 Addenda and was later supplemented by a proposed Technical Specification revision provided by my letter of June 21, 1977.

Additional information concerning steam generators at Oconee has been provided on several occasions. My letter of August 26, 1977, provided a status report of the investigation conducted with regard to steam generator tube leaks. In a letter dated September 9, 1977 a safety assessment of steam generator tube leakage was provided. In response to an NRC Safety Evaluation dated October 4, 1977, which contained commitments for additional information, my letter of December 16, 1977 provided the results of metallurgical examination on two removed tubes, an evaluation of the plugging limit criteria, and a discussion of the inservice eddy current inspection calibration standard. Further information in response to the commitments was provided in a letter dated June 8, 1978. In response to an additional request for information dated December 9, 1977, my letter of February 16, 1978 provided an operating history of steam generators at Oconee.

Recent history has shown a significant reduction in the number of steam generator tube leaks at Oconee and resultant increase in unit availability.

Recent discussions with the staff have indicated that the submittal of another specification on steam generator inservice inspection, similar in content to that approved for another licensee, be provided. Therefore,

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in response to this request and pursuant to 10CFR50, §50.90, the attached proposed Technical Specification is provided for the staff's review and approval. As this proposal supplements earlier submittals in which license fees were attached no further fees are provided. The inservice inspection portion of this submittal has been prepared and modeled on the 1977 Edition of the ASME Code and a similar specification provided for another licensee. The proposal also includes a revision to Specification 3.1.6 to allow a specific action period of seven days in the event the specified leak rate limit of 0.3 GPM for Unit 1 is exceeded. Upon reaching a 1 GPM leak rate or at the end of seven days, whichever occurs first, a reactor shutdown shall be initiated within four hours. This provides a degree of flexibility in the operation of Unit 1, to allow limited operation with a small tube leak. By allowing Unit 1 to continue to operate with an indicated tube leak of up to 1 GPM for up to seven days, significant savings in personnel exposure could be achieved by eliminating unit shutdowns with tubed leaks as small as 0.3 GPM that are difficult and time consuming to locate. The health and safety of the public is maintained while radiation exposure to personnel involved is minimized.

Your prompt attention to this proposal is requested.

Very truly yours,

William O. Parker, Jr.

RLG:scs
Attachment

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47 and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr.

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 23rd day of January, 1979.

Vivian B. Rohlins

Notary Public

My Commission Expires:

February 15, 1982

ATTACHMENT 1

OCONEE NUCLEAR STATION

PROPOSED TECHNICAL SPECIFICATION REVISION

1.8 STEAM GENERATOR TUBING SURVEILLANCE

1.8.1 Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Any eddy current testing indications are considered as imperfections.

1.8.2 Degradation means a service-induced metal loss (e.g. wastage, wear, erosion or general corrosion) or cracking occurring on either inside or outside of a tube.

1.8.3 Degraded Tube means a tube containing imperfections >20% of the nominal wall thickness caused by degradation.

1.8.4 % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

1.8.5 Defect means a degradation of such severity that it exceeds the plugging limit.

1.8.6 Plugging Limit is a degradation of 40%.

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shut down within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the ractor shall be shutdown within 24 hours of detection.

3.1.6.3 If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shut down, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 a. Leakage through the Unit 1 steam generator tubes shall not exceed 0.3 gpm.

b. If the leakage exceeds 0.3 gpm, but is less than 1.0 gpm, unit operation may continue for a seven day period, during which time an evaluation of the leak shall be performed.

c. If the leakage increases above 1.0 gpm, or exceeds 0.3 gpm for seven days, a reactor shutdown shall be initiated. The unit shall be in a cold shutdown condition within the next 36 hours.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is resolved.

3.1.6.8 When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be operable, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means to detect leakage are operable.

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7 except that such losses when added to leakage shall not exceed 30 gpm.

Bases

Every reasonable effort will be made to reduce reactor coolant leakage including evaporative losses (which may be on the order of .5 gpm) to the lowest possible rate and at least below 1 gpm in order to prevent a large leak from masking the presence of a smaller leak. Water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to inplant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small breaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one high pressure injection pump and makeup would be available even under the loss of off-site power condition.

Leakage through steam generator tubes is indicated by increased gaseous activity at the discharge of the unit's air ejectors. The approximate magnitude of the leakage is obtained by a correlation of activity at the air ejector to primary coolant gaseous activity. A radiation monitor on each main steam line indicates the probable generator which has the leak.

If the leakage is to the reactor building, it may be identified by one or more of the following methods:

a. The reactor building air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are .10 gpm to greater than 30 gpm, assuming corrosion produce activity and no fuel cladding leakage. Under these conditions, an increase in coolant leakage of 1 gpm is detectable within 10 minutes after it occurs.

b. The iodine monitor, gaseous monitor and area monitor are not as sensitive to corrosion product activity.[1] It is calculated that the iodine monitor is sensitive to an 8 gpm leak and the gaseous minotor is sensitive to a 230 gpm leak based on the presence of tramp uranium (no fission products from tramp uranium are assumed to be present). However, any fission products in the coolant will make these monitors more sensitive to coolant leakage.

c. In addition to the radiation monitors, leakage is also monitored by a level indicator in the reactor building normal sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as reactor coolant system, low pressure service water system, component cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The sump capacity is 15 gallons per inch of height and each graduation on the level indicates ½ inch of sump height. This indicator is capable of detecting changes on the order of 7.5 gallons of leakage into the sump. A 1 gpm leak would therefore be detectable within less than 10 minutes.

d. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, coolant temperature, pressurizer water level and letdown storage tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a tank level decrease. The letdown storage tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 1 inch of tank height. This inventory monitoring method is capable of detecting changes on the order of 31 gallons. A 1 gpm leak would therefore be detectable within approximately one half hour.

As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on 2 different principles, i.e., activity, sump level and reactor constant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

The upper limit of 30 gpm is based on the contingency of a complete loss of station power. A 30 gpm loss of water in conjunction with a complete loss of station power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore electrical power to the station and makeup flow to the reactor coolant system.

REFERENCES

FSAR Section 11.1.2.4.1

4.17 STEAM GENERATOR TUBING SURVEILLANCE

Applicability

This specification applies to the surveillance of the steam generator tubing.

Objective

To define the in-service surveillance program for steam generator tubing.

Specification

4.17.1 Examination Methods

In-service inspection of steam generator tubing shall include non-destructive examination by eddy current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect tube imperfections in excess of 20% of the minimum allowable as-manufactured tube wall thickness .

4.17.2 Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the plugging limit or containing throughwall cracks shall be removed from service (e.g., plugging, stabilizing).

4.17.3 Selection and Testing

The selection and testing of tubes shall be made on the basis of the following:

a. The examination may in either one or both steam generators.

b. The first sample of tubes selected for each in-service inspection shall include:

   1. Group I - All tubes within three rows of the open inspection lane.

   2. Group II - All in-service tubes, excluding tubes in Group I, with throughwall indications greater than 20% by in-service examination during previous examinations.

   3. Group III - A 3% random sample of the total number of tubes in one steam generator, in addition to those tubes in Groups I and II examined.

c. Subsequent tube examinations will be based on the results of the examinations of Groups II and III tubes only and will be conducted as follows:

   1. If the sum of the number of Group II tubes examined with growth in excess of 10% and the number of Group III tubes

examined with degradation in excess of 20% is less than 5% of the tubes examined and no tubes examined are defective, then no further examinations are required.

2. If the sum of the number of Group II tubes examined with growth in excess of 10% and the number of Group III tubes examined with degradation in excess of 20% is between 5% and 10% of the tubes examined or any tube examined is defective, then plug any defective tube and examine an additional 3% sample of the tubes in the steam generator.

3. If the sum of the number of Group II tubes examined with growth in excess of 10% and the number of Group III tubes examined with degradation in excess of 20% is greater than 10% of the tubes examined or more than 1% of the tubes examined are defective, then plug the defective tubes and inspect an additional 3% of the tubes in the affected generator and inspect a 3% sample of the tubes in the other generator.

4.17.4 Inspection Frequencies

The required inservice inspection of steam generator tubes shall be performed at the following frequencies:

a. The inservice examination of steam generator tubing shall be performed during 40 month periods except as follows:

   1. If, in the examination of any steam generator tubing group more than 10% of the examined tubes in the group indicate any growth of previous degradation (Groups I, II) or new degradation in excess of 20% throughwall (Group III), then the next two inservice examinations will be performed at 18 ± 6 month intervals.

   2. If, in these subsequent examinations, less than 10% of the examined tubes either indicate additional degradation (greater than 10% of the wall thickness) or are tubes with degradation in excess of 20% of wall thickness, then the inspection interval may continue at 40 months.

b. Additional inservice examinations shall be performed during the shutdown subsequent to any of the following conditions:

   1. A seismic occurrence greater than the operating basis earthquake.

   2. A loss of coolant accident requiring engineered safeguard actuation of the low pressure injection system.

   3. A main steam line or feedwater line break.

4.17.5 Reports

a. Following each inservice examination of steam generator tubes, the number of tubes removed from service due to defects in each steam generator shall be reported to the Director, Office of Inspection and Enforcement, Region II within 30 days.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes removed from service.

Bases

The program of periodic in-service inspection of steam generators provides the means to monitor the integrity of the tubing and to maintain surveillance in the event there is evidence of mechanical damage or progressive deterioration due to design, manufacturing errors, or operating conditions. In-service inspection of steam generator tubing also provides a means of characterizing the nature and casue of any tube degradation so that corrective measures may be taken.

In-service inspection includes non-destructive examination using a suitable eddy current inspection system, or other equivalent techniques, capable of locating and identifying degradations due to stress corrosion cracking, mechanical damage, chemical wastage or other causes.

Removal from service will be required for any tube with service-induced metal loss in excess of 40% of the tube nominal wall thickness or with a crack. Additional corrective actions may be required to stabilize a cracked tube.

The initial sample of tubes inspected in a steam generator includes tubes from three different groups. All lane tubes are inspected to assure their integrity. All other inservice tubes with degradation, inspected in previous inspections, are inspected to assure tube integrity and determine degradation growth, if any. Finally, a 3% random sample of tubes is inspected. The results of the latter inspection dictate the extent of further examinations.