NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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DOCKET NUMBER: 50-269/270/287

FILE NUMBER:

TO: Mr Edson G. Case

FROM: Duke Power Company
Charlotte, North Carolina
William O. Parker, Jr.

DATE OF DOCUMENT: 6/21/77

DATE RECEIVED: 6/27/77

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DESCRIPTION

ACKNOWLEDGED

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PLANT NAME: Oconee Units 1-2-3 (4-P)

RJL 6/30/77

ENCLOSURE

Consists of proposed Technical Specification ...notorized 6/21/77....incorporating provisions of the model specifications excluding steam generator tube sample size and frequency......

(7-P)

FOR ACTION/INFORMATION

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DUKE POWER COMPANY
POWER BUILDING
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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

June 21, 1977

Regulatory File Cy

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

Dear Sir:

In response to a July 18, 1974 letter from Mr. Karl R. Goller, a submittal dated August 30, 1974 was made which requested a revision to the Oconee Nuclear Station Technical Specifications to make provisions for surveillance of steam generator tubing. This proposal was consistent with Regulatory Guide 1.83. At the time of this submittal, no inservice inspections of the "once-through" design of steam generators had been performed. In a February 27, 1975 letter, the preliminary results of the first inservice inspection of Oconee 1 steam generator tubing were provided which indicated that the generators were exceptionally clean and free of detectable defects. Also, it was pointed out that the design of the once-through steam generators was significantly different than that of the recirculation type because:

1. The use of full flow condensate polishing with volatile chemical additions reduced the susceptability of the steam side to corrosion attack.

2. Solids such as sodium phosphate are not added.

3. The straight tube, straight shell configuration and broached tube support plate design minimize possible locations of stagnation.

4. The once-through mode of operation eliminates bulk concentrations of impurities.

5. Testing has been performed to assure freedom from vibration damage.

Additionally, it was stated that Duke Power Company was re-evaluating its program for steam generator tubing surveillance to properly reflect the design and operating features of the once-through steam generator and to incorporate experience gained during the Oconee 1 steam generator inspection.

In a March 27, 1975 letter, a detailed description of the first eddy-current inspection results was provided. This inspection was performed in accordance with the methods described in Regulatory Guide 1.83.

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In a September 3, 1975 letter, it was stated that the criteria for steam generator surveillance were being developed for inclusion in Section XI of the ASME Boiler and Pressure Vessel Code. It was intended that this criteria would be issued in the Winter 1975 Addenda to Section XI. Submittal of a specification covering this surveillance was to be provided after the Section XI code was revised to cover this matter.

In a September 21, 1976 letter, Mr. A. Schwencer provided a model technical specification covering steam generator surveillance and requested that we propose a license amendment to incorporate the requirements concerning the operability and inservice inspection of the steam generators. This was accomplished in our submittal of November 30, 1976 in which a Technical Specification modeled on the ASME Section XI Winter 1975 Addenda was provided.

Over the past three years since the original submittal of a Technical Specification covering steam generator operation and surveillance, considerable experience has been gained on the six Oconee once-through steam generators. Regularly scheduled inservice inspections have been performed four times (two for Oconee 1 and one each for Oconee 2 and 3), and have examined approximately 4000 tubes. As a result of steam generator tube leaks which have occurred since July, 1976, nine additional inspections have been performed which have examined approximately 700 additional tubes (the examinations performed during the first several tube leak outages were limited in scope). Additionally, inservice inspections have been performed at two other nuclear stations which utilize the B&W designed once-through steam generators.

It has been our experience that the once-through steam generators are not subject to widespread general tube wall thinning due to corrosive attack. Indeed rarely, if ever, has a tube been observed to have wall thinning in the classical sense. Examinations performed during the tube leak outages have revealed that the leaks are the result of the propagation of a local defect by high-cycle fatigue from vibration. There has been no evidence of intergranular stress corrosion or evidence of the wasting or tube denting problems associated with some of the recirculating steam generators. Eddy current testing has been shown effective in locating a defect in a failed tube, but has been unable to predict a pending failure in a tube. For instance, tubes examined by eddy current testing and shown to be acceptable for continued service have failed within a matter of weeks. In short, eddy current testing has not proven itself to be entirely useful as a diagnostic aid in ascertaining the condition of the steam generator.

Currently, there are at least three recognized inservice inspection programs for steam generator tubes. These are Regulatory Guide (RG) 1.83, the NRC Model Technical Specifications and the Winter 1975 Addenda to ASME Section XI. Based upon our experience, it is our belief that these programs are devised to discover the general degradation which has been identified in certain recirculating water steam generators. The following is a list of comments which Duke Power Company considers pertinent with regards to these standards.

1. All three standards require additional unscheduled inservice inspections during the shutdown subsequent to a primary-to-secondary tube leak in excess of the Technical Specification limits (1 gpm). In the type leaks which have occurred at Oconee (circumferential cracks), the majority of leaks will progress to the technical specification limit. This is due to the necessity of assuring a leakage rate of sufficient magnitude to be identified after shutdown before the shutdown is initiated, and the rapid progression of the leakage rate during shutdown. Since in all instances only one tube has been involved, it is unnecessarily restrictive to require the inspection of 450 to 900 steam generator tubes during every tube leak outage. Indeed, there is no rationale for requiring an inspection based upon leakage rates when the type of leaks encountered have been cracks and not wall thinning.

2. Regulatory Guide 1.83 requires a supplemental inspection of the second steam generator if any tube indicates a previously undetected imperfections of 20% of greater depth. Several eddy current indications have been observed on Oconee tubes in the tube support sheet or tube sheet areas where interpretation of results is difficult. In selected instances, these tubes have been plugged as a precaution. Eddy current testing can accurately tell the wall thickness of defects for wastage and not for cracks. Therefore, it is likely that additional steam generators would be required to be inspected unnecessarily.

3. The model Technical Specification requires excessively large eddy current samples. For instance, if one tube defect is determined during a tube leak outage, a minimum of approximately 2700 tubes must be inspected. This is unreasonably restrictive in restoring the unit to service particularly considering the dubious value of the inspection for once-through steam generators.

In summary, while it is considered that eddy current examination techniques are valid for detecting general steam generator tube wastage (thinning), and hence pending failures, it is not particularly effective for the type of steam generator tube defects which have occurred on the Oconee units. Eddy current examination should be continued, therefore, on such a frequency as would be conducive to detecting and identifying any potential general tube wastage situation. It is not meaningful or beneficial, however, to apply eddy current examination extensively in all cases of tube defects in once-through steam generators. In this regard, it is also noted that the matter of the tube defects which have occurred at Oconee is the subject of separate discussions with the staff.

Accordingly, pursuant to the provisions of 10CFR50.90, the attached proposed Technical Specification is provided for your review and approval. This specification incorporates provisions of the model specifications excluding steam generator tube sample size and frequency. The sample size and frequencies proposed are in accordance with the requirements of ASME Section XI Winter 1975 Addenda. As described above, the requirement to perform an unscheduled inspection in the event a tube leaks in excess of the Technical

Specification limits have been deleted, and it is considered that the sample sizes and acceptance criteria are appropriate for the once-through steam generators.

The Oconee steam generators are, and will continue to be inspected in accordance with the proposed Technical Specifications.

Very truly yours,

William O. Parker

William O. Parker, Jr.

MST:ge

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 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

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 21st day of June, 1977.

Vivian B. Rokkens

Notary Public

My Commission Expires:

Feb. 15, 1982

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown 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 reactor 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 shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If reactor coolant system leakage exceeds 1 gpm through the steam generator tubes, a reactor shutdown shall be initiated within 4 hours and the reactor 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 otherwise corrected.

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 in-plant 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.

If 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 product 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 monitor 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 1/2 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

Applies to the steam generator tubing surveillance.

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 defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.17.2 Selection and Testing

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

a. The examination may be from one steam generator and shall include 1 1/2% of the total installed steam generator tubes for the unit.

b. Every inspection shall include all tubes which previously had detectable wall penetrations (greater than 20 percent and not including plugged tubes), and shall also consider tubes and those areas where design and experience have indicated potential problems.

If the inspection indicates more than 10 percent of the inspected tubes have detectable wall penetrations (greater than 20 percent), an additional inspection encompassing 3 percent of the tubes in both steam generators shall be examined, concentrating on areas of the tube array where the tubes with defects were found. In the event the inspection of these additional tubes indicates that more than 10 percent of the tubes examined in any single steam generator have detectable wall penetrations (greater than 20 percent), an additional inspection encompassing 3 percent of the tubes in both steam generators shall be examined.

4.17.3 Inspection Intervals

a. The inservice examinations of steam generator tubing shall be performed during 40 month periods except as indicated in Specification 4.17.3.b and 4.17.3.c.

b. If in any examination of steam generator tubing an excess of 10 percent of the tubes exhibit indications in excess of 20 percent of the wall thickness, the next two inspections shall be performed at 12 to 24 month intervals. If, in these examinations, no more than 10 percent of the tubes examined exhibit either additional degradation (greater than 10 percent of wall thickness) of previously degraded tubes, tubes with new defects in excess of 20 percent of wall thickness, or a combination of both, the inspection interval may continue at 40 months.

c. 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 actuation of the engineered safeguards system,

   3) a main steam line or feedwater line break.

4.17.4 Acceptance Criteria

a. If less than 10 percent of the total tubes inspected have detectable wall penetration (greater than 20 percent), operation may resume after required corrective measures have been taken.

b. If more than 10 percent of the total tubes inspected have detectable wall penetrations (greater than 20 percent), operation may resume after required corrective measures have been taken, and the situation and remedial action shall be reported to the NRC.

4.17.5 Corrective Measures

All tubes with unacceptable defects (greater than 40 percent wall thinning) shall be plugged.

4.17.6 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the NRC within 30 days.

b. The results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this 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.

Bases

The program of periodic in-service inspection of steam generators provides the means of monitoring 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 cause 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 defects due to stress corrosion cracking, mechanical damage, chemical wastage, or other causes.

An unacceptable defect is defined as one which would result in not satisfying the calculated acceptable minimum tube wall thickness that can sustain a loss-of-coolant accident in combination with a safe shutdown earthquake.