NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-269/270/287

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr. A. Schwencer

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

DATE OF DOCUMENT 5/11/77

DATE RECEIVED 5/16/77

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DESCRIPTION

Ltr. trans the following:

(1-P)

PLANT NAME:

Oconee Units 1-2-3

RJL

ENCLOSURE

Consists of requested additional information to allow assessment of the Oconee steam generator tube leak occurrences......

ACKNOWLEDGED

(8-P) DO NOT REMOVE

FOR ACTION/INFORMATION

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NRC FORM 195 (2-76)

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

May 11, 1977

RECEIVED
MAY 16 1977
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

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

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch #1

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

Regulatory File Cy.

Dear Sir:

Your letter of April 5, 1977 stated that certain additional information was necessary to allow your assessment of the Oconee steam generator tube leak occurrences. This information is attached per your request.

Very truly yours,

William O. Parker, Jr.

MST:vr

Attachment

771360233

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RELATED TO OCONEE STEAM GENERATORS

QUESTION 1

It was stated during the meeting on February 15, 1977, that defective and plugged tubes were stabilized down to the top support plate if the defect was found near the top tube sheet. Assess the consequence of possible failures of these defective tubes at lower or un-stabilized sections.

RESPONSE:

The tube stabilization procedure prevents the possibility of primary to secondary system leakage. A stabilized tube is plugged at the lower tube sheet using conventional techniques. The stabilizer end fitting is welded into place and serves as the tube plug at the upper tube sheet elevation. This plugging process isolates the generator tube from primary system flow. Consequently, a postulated failure of a stabilized tube would not result in a primary to secondary system leak.

The tube stabilizer being approximately 109 inches long, extends through the 14th support plate by 2 to 3 inches. Secondary side flow below the 14th support is characterized as having no cross flow components. (In fact, even that region between the 15th and 14th support plates experiences little, if any, cross flow). As a result, that portion of the generator tube which has not been stabilized should not experience any significant excitation. Therefore, the failure of a tube below the stabilizer is unlikely.

QUESTION 2

Provide a re-evaluation of past ECT records to show whether or not there were tube defects that might have led to initiation of tube cracking.

RESPONSE:

Eddy-current data taken during both normal inservice inspection and tube leak outages have been reviewed and evaluated by B&W. It has been our experience that although the Eddy-Current technique is very useful for identifying tube abnormalities during rapid scanning, it will not totally characterize a flaw with respect to its type, shape, orientation or potential for failure. With specific regard to the possibility of identifying impending failures at Oconee, our experience to date would indicate that no direct correlation between eddy-current indication and tube failures exists at this time. Because of this limitation, and our own needs for information of this nature, a decision was made to obtain a generator tube sample which would provide greater insights into this matter.

Due to the history of the recent tube failures at the Oconee Site, B&W's efforts have been initially concentrated in the "open tube lane" area of the generator at an elevation which is between the 15th support plate and the upper tube sheet. It is in this area that results from eddy-current examinations indicate tube-to-tubesheet and tube-to-tube-support plate responses which differ from those generally observed in the tube bundle. Because of this distinction and the incidence of "lane tube" failures, tube 77/25 from the Oconee 1-B generator was removed for detailed laboratory examination to ascertain the true significance of the eddy-current indications.

The results of this examination have not yet been finalized. Detailed investigations and evaluations are currently underway to clarify any anomalies and to determine if they could possible relate to the tube failure problems being encountered. Some lab simulation and comparison work has been completed. On this basis, it appears that the upper tube sheet signal being detected is caused by a slight groove or indentation, ($\approx$1.0 mil.). in the tube at the O.D. surface.

QUESTION 3

Due to failure in all the affected units at nearly the same point in time, indicate any change in operating procedures or other possible incidents that might have led to tube failures in Oconee steam generators.

RESPONSE:

Reviews of current operational practices have been conducted to determine if they could possibly be causing or contributing to recent OTSG tube failures. No definite correlations have been noted. However, one item has been identified as possibly being related to tube failures, the increased frequency of testing turbine stop and control valves. Prior to July, 1975, the turbine stop and control valves were tested monthly. Since that time, these components have been tested daily and weekly respectively in accordance with the vendor recommendation. Recent tests have confirmed that the cycling of these components induces pressure transients in the steam generators.

Further tests and analyses of the possibility of this being the cause of tubing failures are in progress. Presently, the frequency of testing of turbine stop and control valves have been reduced to monthly, and is being performed at reduced power, in conformance with Oconee technical specification until the evaluation is completed.

QUESTION 4

Indicate any plan to perform ECT examinations of periphery tubes.

RESPONSE:

For the six generators at the Oconee site, two leaking tubes (tubes 114/109 and 32/13 of Unit 1B) have been experienced which were not on the "open lane". Both of these tubes are within 6 tubes of the periphery of the tube bundle. Each has been "stabilized".

Other significant experience with tubes in the "peripheral region" (within approximately 10 tubes of the periphery) would be the plugging or stabilization of tubes 113/110, 33/14, 2/7, 2/8, and 101/4 which were non-leakers but which had significant eddy-current indications.

Normal inservice inspection has been performed on all generators at Oconee. Specifically, two ISI's on Oconee Unit 1 and one each of Units 2 and 3 have been conducted. This constitutes the inspection of approximately 3,600 tubes. Of these, roughly 1000 should have been within the "peripheral region" of the tube bundle.

In the course of investigating the leak of tube 31/13, eddy-current analysis of approximately 100 additional tubes in the "peripheral region" were conducted. Further investigation of the peripheral area will be conducted as considered appropriate.

QUESTION 5

Provide analytical calculations and/or tests to justify that the crack length, in the circumferential directions, associated with the proposed leakage rate will not increase in an unstable fashion under normal operating and accident conditions.

RESPONSE:

All of the defects which have been visually observed at the Oconee site have been circumferential cracks of varying length. One of these failed tubes, tube 77/23 of generator 2B, was removed for detailed examination. Evidence obtained from the fracture surface of this tube indicates that from an initiation site of unknown origin the crack propogated as a thru-wall defect due to the application of a high cycle fatigue loading. It was deduced that approximately 1×10^5 to 3×10^5 cycles were required for the crack to travel its total observed length of roughly 240°. The only known source of loading which would involve this number of cycles is flow induced virbration. This would occur with most prominence in the fundamental mode of the tube, or at a frequency of about 40 Hz. From this information, it is apparent that an initial defect would propogate to a detectable leak in approximately 1 to 2 hours.

Since the propogating mechanism is flow induced vibration, the defect is not "stable". It will rapidly progress around the curcumference of the tube as long as there is flow of sufficient energy to drive it. However, the crack formed will produce an identifiable leak and the unit will be shut down promptly. Therefore, the probability of the occurence of a major accident during the time between leak and shutdown is low.

QUESTION 6

During the recent meeting with the NRC staff, it was indicated that there is 0.4% sulphur content in the sediment deposits. Provide an assessment on the effect of the high sulphur content to the tubes in terms of possible chemical reactions.

RESPONSE:

As part of the overall examination plan being conducted, chemical analysis was performed on deposit samples removed from the tube surfaces. The results of these analyses revealed the deposit to be primarily iron oxide as magnetite. Other elements were present only as minor constituents of the deposits with no deleterious amounts of contaminants noted. The analyses revealed sulfur levels in the deposits typically to be below the 0.2% detection limit with one value reported to be 0.4%. These sulfur contents in the CRUD are lower that typical sulfur levels measured in as-fabricated vessels. The sulfur levels in as-fabricated units have been shown to be acceptable over a wide range of operating conditions. Thus, the levels of sulfur found in the tube deposits are not considered consequential in terms of tube corrosion.

A review of the operating environment further alleviates concerns for the low sulfur levels present in the deposit. Operating experience has indicated sulfur to be a problem with Alloy 600 tubing at low temperatures, primarily under improper wet layup conditions in steam generators. At these lower temperatures sulfur can be present in the reduced species which can produce intergranular attack of Alloy 600 under certain conditions. Proper wet layup chemistry control, as specified for the Oconee steam generators, provides sufficient control to avoid this potential corrosion problem. Additionally, there has been no evidence of sulfur induced corrosion of Alloy 600 steam generator tubes at operating temperatures. Such corrosion is not expected with the alkaline pH levels maintained in the feedwater and the imporbability of reduced sulfur species being present at operating temperatures.

Based on the above, the sulfur levels seen in the tube deposits are not of concern because of the low levels present and once through steam generator operating conditions.

QUESTION 7

Provide the micro-hardness test results of both virgin and cracked tubes to determine any evidence of plastic cyclic straining that may initiate the cracks.

RESPONSE:

B&W has performed a significant amount of micro-hardness testing of virgin Alloy 600 tubing as part of their original OTSG materials evaluation program. There is an inherently large amount of scatter in micro-hardness testing because of the localized nature of the test. Results, therefore, are best evaluated in terms of average results. In all, 27 different tubes were evaluated in these tests. The tubes tested were taken from production tubes and are representative of those used in the Oconee once-through steam generators. Results on the virgin tubes reveal knoop hardness values ranging from 165 to 222.

Micro-hardness tests were recently performed on the two tube samples, tubes 77/23 and 77/27 which were removed from Oconee 2B generator for detailed examination. On tube 77/23, the tests were run on a specimen approximately 1/2 inch up from the fracture surface. On tube 77/27, the tests were run on a specimen which would be located approximately 8 inches down from the tube to tube sheet interface. Sets of measurements on a transverse section of the specimen were taken as a radial function of position at 5 mils from the inside surface, at the center of the tube wall and at 5 mils from the outside surface. Five azimuthal locations were selected; 0°, 60°, 120°, 180° and one randomly selected angle. Also, an additional set of measurements were taken across the tube wall of a longitudinal section of the specimen. The results of this investigation are provided in Table I. As can be seen, the hardness does not vary appreciably between tubes 77/27 and 77/23 at the locations tested.

It should be noted that the results tabulated in Table I are fickers hardness numbers. Although it may not be technically correct to compare knoop and fickers hardness data directly, the method utilized in each of these procedures is sufficiently close to justify a comparison for our purposes at this time. The results show that the data taken from virgin specimens compares well with that taken from tubes 77/23 and 77/27.

B&W is currently planning to perform additional micro-hardness tests on other pieces of the Oconee generator tube samples. This more extensive test program will provide data which is directly comparable and should be vital in establishing wheter or not any local plastic cycling has occurred.

QUESTION 8

Provide accident consequence analyses assuming:

a. A certain number of tube failures, that can be tolerated, concurrent with a LOCA.

b. The equivalent number of tubes failures that can be tolerated during a MSLB in terms of off site dosage.

RESPONSE:

A study has been made of the environmental consequence of a steam line break accident followed by the rupture of a steam generator tube such that a large primary-to-secondary leak rate (640gpm) exists in the affected staam generator. The 640 gpm leak rate is at reactor operating temperature and is approximately equivalent to the Oconee FSAR leak rate of 435 gpm at the density for cold conditions. For this analysis it is assumed as in the Oconee FSAR Section 14.1.2.9, that the reactor coolant leakage continues unabated for three hours before the reactor coolant system can be cooled down and the leakage terminated. In evaluating the environmental consequences, the Oconee FSAR Section 2.3 meterology was used, that is for a ground level release from 0 to 2 hours, the atmospheric dispersion factor (X/Q) at the exclusion area boundary is 1.16×10^{-4} sec/m^3. The reactor coolant system iodine inventory is based on 1.0% defective fuel and the source terms from the Oconee FSAR Section 11.1. The reactor coolant iodine concentration is 4.6 μCi/cc of dose-equivalent I-131. The thyroid dose at the exclusion area boundary was calculated as follows:

The steam generator tube rupture causes all of the iodine activity in the entire reactor coolant to be released through the steam line break directly to the atmosphere. It was assumed that there was no further release of iodine from the fuel into the reactor coolant as a result of the steam line break transient. It was also conservatively assumed that the entire iodine release occured over a two-hour time period. The resulting thyroid dose at the exclusion area boundary is 91 rem. Therefore, the entire reactor coolant volume with an iodine inventory corresponding to 1.0% defective fuel can be released directly to the atmosphere via ruptures in one or more steam generator tubes following a steam line break since the resulting thyroid dose is well below the 10CFR100 guideline of 300 rem.

B&W has no approval procedures and methods for calculating the consequences of a LOCA with a steam generator tube rupture. Conservative hand calculations have been performed which estimate that offset failure of three steam generator tubes would result in minimal impact on peak cladding temperature calculations. The effect of three tube failures on peak containment pressure would be insignificant.

In the LOCA analysis performed to show compliance of the ECC systems to 10CFR50.46 for the Oconee plants, no credit was taken for steam flow through the loops during the reflooding phase of the transient. It has been postulated that a loop seal may occur in the pump suction piping that will prevent loop venting. However, calculation performed with the CRAFT code show that no loop seal is present at the end of blowdown. If loop venting was used in the reflooding analysis, flooding rates would increase 70% over the valves used to demonstrate compliance to 10CFR50.46. Therefore, if a realistic calculation of the reflooding phase was performed for the Oconee plants, offset rupture of 20 or more tubes could probably be tolerated without affecting the present Oconee LOCA limits.

QUESTION 9

Provide analytical and/or test data to assure tube integrity by demonstrating the capability of degraded tubes (circumferentially partial cracked tubes) to withstand accident induced loads. NRC's positions on this matter were delineated in Regulatory Guide 1.121 which was published for comment in August 1976.

RESPONSE:

As has been previously established, refer to Item 5, a small defect induced in the tube rapidly propagates by flow induced vibration to a crack with a detectable leakage rate. This "fast break" phenomenon and resultant plant shutdown procedure constitutes sufficient assurances that the chances of a "degraded" tube being subjected to accident loading conditions is low.

Small defects which are in the process of forming but which has not yet propogated to a detectable leak do not substantially affect the gross structural integrity of the tube. Consequently, loads induced during an accident condition should not cause a tube which is in this particular state to fail.

QUESTION 10

Indicate B&W's on-going and planned future programs associated with tube failute, i.e., tests on mechanical strengths and fatigue strengths of degraded tubes.

RESPONSE:

On February 15, 1977, a meeting was held in Bethesda, Md. between representatives of the NRC, Duke Power and B&W. At that time, the status of the tube leak problem at Oconee Units 1, 2 and 3 was discussed at length. Involved in that discussion was a presentation which related B&W's current and on-going programs associated with the resolution of the tube failutes. Results presently at hand were provided for the detailed visual, chemical and metallographic investigations being conducted as part of B&W's total investigative study.

With regard to future plans, Duke Power and B&W identified that at the first available opportunity, another generator tube would be removed for detailed examination. This operation has been completed and our investigations are currently underway. Specific areas such as mechanical and fatigue strength tests on actual tube samples are being contemplated. As sample data and design information currently exists in the literature with respect to these properties, this type of test information is not viewed as being imperative at this time. However, these tests would provide confirmatory type data which could then be used as verification for the present "design" information.

TABLE 1

MICROHARDNESS RESULTS

Specimen #	Orient	Test Location	DPH 0°	DPH 60°	DPH 120°	DPH 180°	DPH Random
27M-1	Trans.	ID	200	229,214	201	122	232
"	"	Center	166,162,178	167	178,170	203	174
"	"	OD	184	168,188	211	170	172
27M-1	Long.	ID					175
"	"	Center					160
"	"	OD					170
23T-3	Trans.	ID	188	188	190	191	194
"	"	Center	212,186	202	169,184,184	191	182
"	"	OD	191	194	214	191	183
23T-3	Long.	ID					197
"	"	Center					169
"	"	OD					183

MAY 5 1977

Dockets Nos. ~~50-269~~
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

By letter dated November 30, 1976, you requested an amendment to the license for the Oconee Nuclear Station to incorporate requirements concerning the operability and inservice inspections of steam generators. Your request was in response to our letter dated September 21, 1976, which provided model Technical Specifications to be adapted to the Oconee Technical Specifications. We find that your response does not address all of the requirements covered in our model Technical Specifications. For those requirements you have addressed, we request additional information.

In our letter of September 21, 1976, we provided model Technical Specifications which included a reactor coolant leakage limit of 1 GPM through the steam generator tubes. Your letter of November 30, 1976, stated without justification, that a 1 GPM leakage rate is overly restrictive and that the 10 GPM now allowed by the Oconee Technical Specifications is adequate. It is our position that this leakage rate should be limited to 1 GPM, particularly in view of the steam generator tube leaks which have been occuring at Oconee and other PWR facilities. It is requested that you submit a request for change to the Oconee Technical Specifications that limits the leakage rate to 1 GPM or provide detailed justification for not doing so.

An analysis performed by us shows that with a 1 GPM reactor coolant-to-secondary leakage rate, dose rates from postulated accidents would be well below the limits of 10 CFR Part 100. This analysis assumed that the reactor coolant activity was 1.0 $\mu ci/_g$ and the secondary coolant activity was 0.1 $\mu ci/_g$.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

We have reviewed the Oconee Technical Specifications and find that they do not include iodine activity limit for the reactor coolant and the iodine limit for the secondary coolant is well above that assumed in our analysis. It is requested that you submit a request for a change to the Oconee Technical Specifications that would limit the iodine activity to 1.0 $\mu ci/_{gm}$ and 0.1 $\mu ci/_{gm}$ in the reactor and secondary coolants, respectively. Enclosure 1 is a copy of the B&W Standard Technical Specifications which you should use for guidance.

Enclosure 2 contains comments we have on your proposed Steam Generator Inservice Inspection Technical Specifications. It is requested that you respond to these comments by modifying your proposed Technical Specifications to conform with the model Technical Specifications provided in our letter of September 21, 1976, or by providing justification for any deviations.

It is requested that you respond to the requests herein within 45 days of receipt of this letter.

Sincerely,

Original signed by

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

Enclosures:
1. B&W Standard Technical Specifications
2. NRC Comments on Steam Generator Inservice Inspection

cc w/encl:
See next page

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

May 5, 1977

Dockets Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

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Sincerely,

A. Schwencer

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

Enclosures:
1. B&W Standard Technical Specifications
2. NRC Comments on Steam Generator Inservice Inspection

cc w/encl:
See next page

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

DEFINITIONS

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT

1.18 QUADRANT POWER TILT is defined by the following equation and is expressed in percent.

QUADRANT POWER TILT =

$$100 \left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right)$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) 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."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY shall be the average (weighted 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.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or designated components at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL POWER IMBALANCE

1.23 AXIAL POWER IMBALANCE shall be the THERMAL POWER in the top half of the core expressed as a percentage of RATED THERMAL POWER minus the THERMAL POWER in the bottom half of the core expressed as a percentage of RATED THERMAL POWER.

SHIELD BUILDING INTEGRITY

1.24 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed.

b. The shield building filtration system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.25 The REACTOR PROTECTION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131.

b. $\leq$ $100/\bar{E}$ μCi/gram.

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

ACTION:

MODES 1, 2 and 3*.

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10% of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/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} < (500)°F within 6 hours.

c. With the specific activity of the primary coolant > $100/\bar{E}$ μCi/gram, be in at least HOT STANDBY with T_{avg} < (500)°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

*With T_{avg} $\geq$ (500)°F.

ACTION: (Continued)

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

2. Fuel burnup by core region.

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

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded.

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

SURVEILLANCE REQUIREMENTS

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

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once each 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\overline{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ μCi/gram.	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

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

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

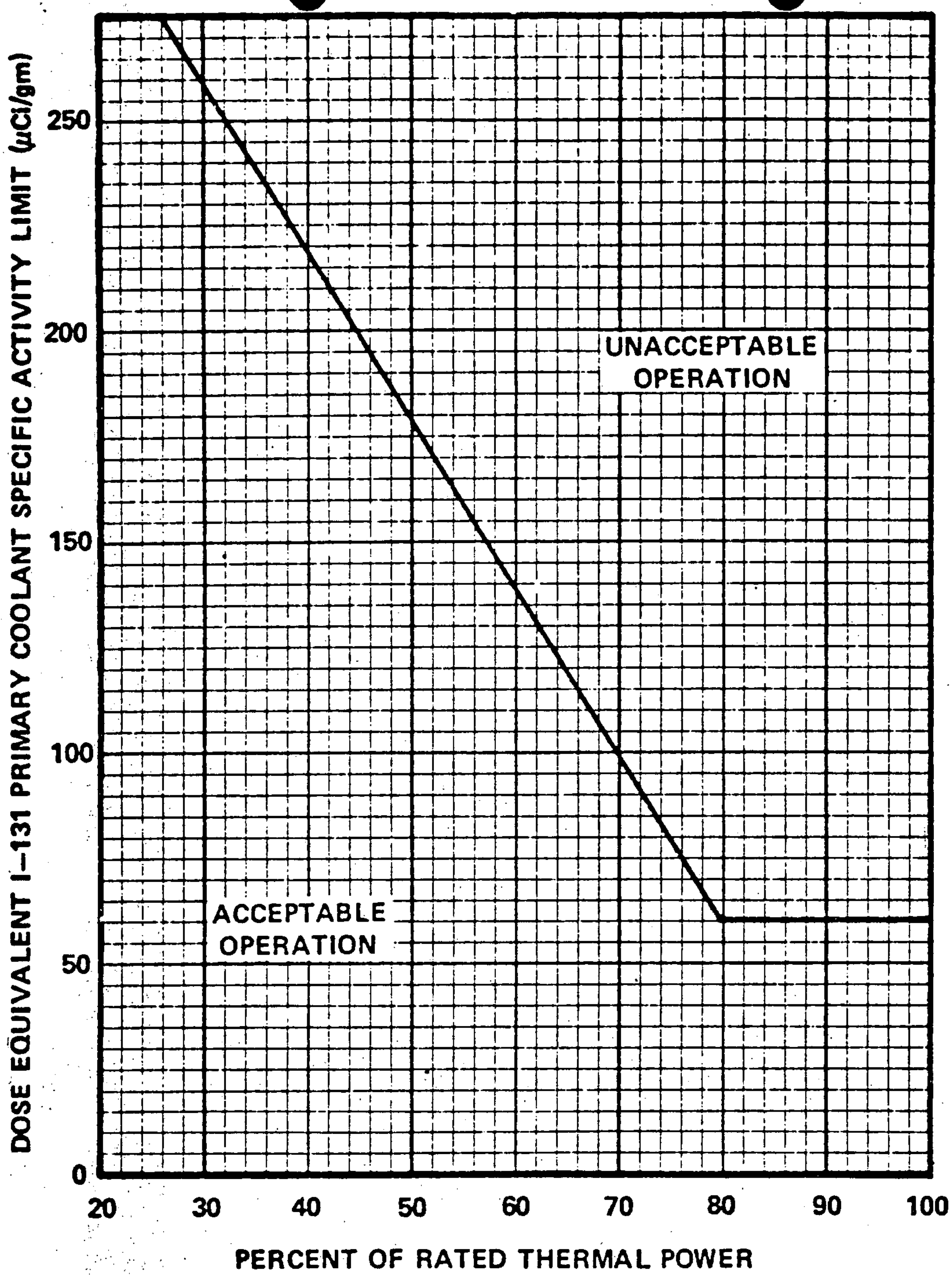

FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

BASES

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits shown on Table 3.4-1 provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 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 the Part 100 limit 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 interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in the specific site parameters of the site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/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. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the units yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < (500)°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (   ) of the FSAR. During heatup and cooldown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

BASES

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEMS

The OPERABILITY of the auxiliary feedwater systems ensures that the Reactor Coolant System can be cooled down to less than (305)°F from normal operating conditions in the event of a total loss of offsite power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of (350) gpm at a pressure of (1133) psig to the entrance of the steam generators. Each steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of (700) gpm at a pressure of (1133) psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than (305)°F where the Decay Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than (305)°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for (   ) hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the

ENCLOSURE 2

COMMENTS ON OCONEE UNITS 1, 2, AND 2 PROPOSED TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR INSERVICE INSPECTION

1. Table 4.17-1 which specifies steam generator tube sample size, inspection result classification, and the corresponding action required for each sample inspection has several discrepancies with the standard technical specifications:

   a. If the results of the first sample inspection fall in the C-2 category, the corrective action should be the plugging of the defective tubes and inspection of $6\frac{N}{n}\%$ additional tubes in that steam generator. Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

   b. If the results of the first sample inspection of a steam generator fall in the C-3 category, the corrective action should include inspection of all tubes in the steam generator, plugging of defective tubes, and inspection of an additional $6\frac{N}{n}\%$ of the tubes in each other steam generator.

   c. Corrective actions corresponding to results of second and third sample inspections have not been specified in the table. These actions should be specified in accordance with Table 4.4-2 of the standard technical specifications. Guidance for subsequent (second and third) sample inspections as stated in paragraph 4.17.1,b of the proposed technical specifications is unacceptable.

   The sample sizes required during each sample inspection are clearly specified in Table 4.4-2 of the standard technical specifications. Deviation from the specified sample sizes will require a statistical analysis justifying the proposed sample sizes.

2. Paragraphs 4.4.5.2,b, and c of the standard technical specifications should be included under paragraph 4.17.1,b of the proposed technical specifications.

3. Paragraph 4.17.2,b should read the same as paragraph 4.4.5.3,b in the standard technical specification.

4. Paragraph 4.17.2,c should also specify that unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection during the shutdown subsequent to:

   a. A seismic occurrence greater than the Operating Basis Earthquake,

   b. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   c. A main steam line or feedwater line break.

5. Justification for the proposed 40% plugging limit must be provided in accordance with Regulatory Guide 1.121.

6. The term "unserviceable" used in the definition of plugging limit must be defined.

7. The definition of defect should read as follows:

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

8. The basis should state:

   a. Cracks having a primary-to-secondary leakage less than the specified limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

   b. Cases when the results of any steam generator tubing inservice inspection fall into category C-3 will be considered by the Commission on a case-by-case basis and may result in a requirement for analyses, laboratory examination, tests, additional eddy current inspection, and revision of the Technical Specification, if necessary.

9. The subject of operability of steam generators as discussed in sections 3.4.5 and 4.4.5.6 of the standard technical specifications should be addressed.