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ATTACHMENT A

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)

2.1.3 Reactor Coolant Radioactivity

Applicability

Applies to the radioactivity of the reactor coolant.

Objective

To ensure that the reactor coolant radioactivity is maintained at a level commensurate with the occupational and public safety.

Specification

(1) The radioactivity of the reactor coolant shall be limited to:

a. $\leq$ 1.0 μCi/gm DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}$ μCi/gm

(2) With the radioactivity of the reactor coolant >1.0 μCi/gm DOSE EQUIVALENT I-131 but $\leq$ 60 μCi/gm, operation, to include restart if shutdown, may continue for up to 100 hours cumulative operating time under these circumstances, not to exceed 800 hours in any consecutive 12 month period.

(3) With the radioactivity of the reactor coolant >1.0 μCi/gm DOSE EQUIVALENT I-131 for more than 100 hours during one continuous time interval or exceeding 60 μCi/gm, be in at least HOT SHUTDOWN with T_{avg} <536°F within 6 hours.

(4) With the radioactivity of the reactor coolant > $100/\bar{E}$ μCi/gm, be in at least HOT SHUTDOWN with T_{avg} <536°F within 6 hours.

(5) With the radioactivity of the reactor coolant >1.0 μCi/gm DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of items 1.(a)(2)(ii) and 1.(b)(2)(i) of Table 3-4 until the radioactivity of the reactor coolant is restored to within its limits.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.3 Reactor Coolant Radioactivity (Continued)

Basis

The limitations on the radioactivity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will be well within the limits of 10 CFR Part 100 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 and a concurrent loss of offsite power.

Permitting power operation to continue for limited time periods with the reactor coolant's radioactivity levels >1.0 μCi/gm DOSE EQUIVALENT I-131, but ≤60 μCi/gm, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing T_{avg} to <536°F prevents the release of radioactivity should a steam generator tube rupture, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive radioactivity levels in the reactor coolant will be detected in sufficient time to take appropriate corrective action(s).

References

USAR, Section 11.11.3

USAR, Section 14.14

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.

(ii) 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 3-8.

e. Reporting Requirements

(i) Following each in-service inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 30 days.

(ii) The complete results of the steam generator tube inservice inspection shall be reported to the Commission within six (6) 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 imperfection.

3. Identification of tubes plugged.

(iii) Results of steam generator tube inspections which fall into Category C-3 require prompt notification of the Commission and shall be reported pursuant to Section 5.6 of the Technical Specifications 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.

(3) Surveillance of Reactor Coolant System Pressure Isolation Valves

a. Periodic leakage testing* on each valve listed in Table 2-9 shall be accomplished prior to entering the power operation

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3-8

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 300 tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 600 tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 1200 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	Inspect all tubes in this S.G., plug defective tubes and inspect 600 tubes in other S.G. Prompt notification to NRC pursuant to Specification 5.6.	C-3	Perform action for C-3 result of first sample	N/A	N/A
			The second S.G. is C-1	None	N/A	N/A
			The second S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			The second S.G. is C-3	Inspect all tubes in the second S.G. and plug defective tubes. Prompt notification to NRC pursuant to Specification 5.6	N/A	N/A

N/A = Not Applicable

3.0 SURVEILLANCE REQUIREMENTS

3.3 Reactor Coolant System, Steam Generator Tubes, and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance (Continued)

Basis

Undetected prolonged leakage of borated reactor coolant onto carbon steel sets up an unusual corrosion mechanism. Detection of this leakage at an early stage can best be accommodated directly after an outage and before startup. The inspection program specified in Specification 3.3(1) places major emphasis on the areas of highest stress concentration as determined by general design evaluation and experience with similar systems. The inspections will rely on non-destructive analysis methods utilizing up-to-date analyzing equipment and trained personnel. Volumetric inspection of the reactor pressure vessel is to be performed completely from the outside diameter. The testing techniques and acceptance criteria of Section XI of the ASME B&PV Code will be utilized, except where specific relief is granted by the Commission.

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 in-service inspection of the steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1, dated July 1975. In-service 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 in-service conditions that lead to corrosion.

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 can be taken.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled in-service steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 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 in-service inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Section 5.6 of the Technical Specifications prior to the 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.

References

(1) USAR, Section 4.5.3

5.0 ADMINISTRATIVE CONTROLS

Responsibilities

5.5.1.6 The Plant Review Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 5.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Manager - Fort Calhoun Station to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Manager - Nuclear Production and to the Chairman of the Safety Audit and Review Committee.

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Safety Audit and Review Committee.

h. Review of the Site Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Safety Audit and Review Committee.

i. Review of the Site Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Safety Audit and Review Committee.

j. Review of all Reportable Events.

Authority

5.5.1.7 The Plant Review Committee shall:

a. Recommend in writing to the Manager - Fort Calhoun Station approval or disapproval of items considered under 5.5.1.6(a) through (d) above.

5.5.2.7 c. Proposed tests or experiments which involve an unreviewed safety question as defined in section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedure or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All Reportable Events.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Plant Review Committee.

The Chairman of the Safety Audit and Review Committee (SARC) may designate subgroups, special working committees, or audit teams as he deems necessary in order to carry out the responsibilities of the SARC. These subgroups, committees, or audit teams will perform the SARC responsibilities and report on their activities for review at the next regularly scheduled SARC meeting following any group's action.

Audit

5.5.2.8 Audits of facility activities shall be performed under the cognizance of the Safety Audit and Review Committee. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per two years.

5.6 Reportable Event Action

5.6.1 The following actions shall be taken in the event of a REPORTABLE EVENT:

a. The Commission shall be notified pursuant to the requirements of 10 CFR 50.72, if applicable.

b. Each Reportable Event shall be reviewed by the Plant Review Committee and submitted to the Chairman of the Safety Audit and Review Committee and the Division Manager - Nuclear Production.

c. Submit reports of Reportable Events pursuant to the requirements of Specification 5.9.2.

5.7 Safety Limit Violation

5.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within six hours.

b. The Safety Limit violation shall be reported to the Commission, the Division Manager - Nuclear Production and to the Chairman of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, Chairman of the Safety Audit and Review Committee and the Division Manager - Nuclear Production within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the minimum requirements of sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33 except as provided in 5.8.2 and 5.8.3 below.

5.8.2 Each procedure and administrative policy of 5.8.1 above, and changes thereto, shall be reviewed by the Plant Review Committee and approved by the Manager - Fort Calhoun Station prior to implementation and periodically as set forth in each document.

5.8.3 Temporary changes procedures 5.8.1 above may be made provided:

5.5.2.7 c. Proposed tests or experiments which involve an unreviewed safety question as defined in section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedure or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All Reportable Events.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Plant Review Committee.

The Chairman of the Safety Audit and Review Committee (SARC) may designate subgroups, special working committees, or audit teams as he deems necessary in order to carry out the responsibilities of the SARC. These subgroups, committees, or audit teams will perform the SARC responsibilities and report on their activities for review at the next regularly scheduled SARC meeting following any group's action.

Audit

5.5.2.8 Audits of facility activities shall be performed under the cognizance of the Safety Audit and Review Committee. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per two years.

5.6 Reportable Event Action

5.6.1 The following actions shall be taken in the event of a REPORTABLE EVENT:

a. The Commission shall be notified pursuant to the requirements of 10 CFR 50.72, if applicable.

b. Each Reportable Event shall be reviewed by the Plant Review Committee and submitted to the Chairman of the Safety Audit and Review Committee and the Division Manager - Nuclear Production.

c. Submit reports of Reportable Events pursuant to the requirements of Specification 5.9.2.

5.7 Safety Limit Violation

5.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within six hours.

b. The Safety Limit violation shall be reported to the Commission, the Division Manager - Nuclear Production and to the Chairman of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, Chairman of the Safety Audit and Review Committee and the Division Manager - Nuclear Production within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the minimum requirements of sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33 except as provided in 5.8.2 and 5.8.3 below.

5.8.2 Each procedure and administrative policy of 5.8.1 above, and changes thereto, shall be reviewed by the Plant Review Committee and approved by the Manager - Fort Calhoun Station prior to implementation and periodically as set forth in each document.

5.8.3 Temporary changes procedures 5.8.1 above may be made provided:

5.9.1 Continued

work and job functions,[3] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report. This monthly report shall also include a statement regarding any challenges or failures to the pressurizer power operated relief valves or safety valves occurring during the subject month.

5.9.2 Reportable Events (LER)

A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to Region IV of the NRC, within 30 days after discovery of the event for the following:

a. Any operation or condition prohibited by the safety limits and limiting safety system settings (Section 1.0) and/or the limiting conditions for operation (Section 2.0) of these Technical Specifications.

b. The occurrence of other events prescribed in 10 CFR 50.73.

[3] This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

5.9.2 Continued

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5.9.2 Continued

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5.9.2 Continued

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5.10 Record Retention

5.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Licensee Event Reports (LER).

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of annual physical inventory of all source material of record.

5.10.2 A complete record of the analysis employed in the selection of any fuel assembly to be placed in Region 2 of the spent fuel racks will be retained as long as that bundle remains in Region 2 (reference Technical Specifications 2.8 (12) and 5.8.4).

DISCUSSION, JUSTIFICATION, AND
SIGNIFICANT HAZARDS CONSIDERATIONS

The Fort Calhoun Station Technical Specifications are being revised in response to Generic Letter 83-43, "Reporting Requirements of 10 CFR 50, Sections 50.72 and 50.73, and Standard Technical Specifications". These changes are in a format similar to the recommendations of Generic Letter 83-43. Included are changes to the Table of Contents and to Sections 2.0, 3.0 and 5.0 of the Fort Calhoun Station Technical Specifications which will ensure continuity of reporting requirements.

The Table of Contents has been changed to reflect the section title changes for sections 5.6 and 5.9.2. Section 5.6 is changed to "Reportable Event Action" from "Reportable Occurrence Action" and Section 5.9.2 is changed to "Reportable Events" from "Reportable Occurrences". These changes are consistent with the recommendations of Generic Letter 83-43. In addition Section 4.5, "Seismic Design for Class I Systems", has been added to the Table of Contents. It was deleted with the issuance of Amendment 86. Item 5.12 has been changed to "Environmental Qualification" from "Environmental Qualifications".

Section 2.1.3 Specification (5) has been changed to remove reportability of reactor coolant Dose Equivalent I-131 being greater than 1.0 μCi/gm. This change is consistent with item (g) of 10 CFR 50.73.

Section 3.3 Specification (2)e(iii) has been changed to reference Section 5.6 for reportability as opposed to referencing Section 5.9.2. The same change has been made in Table 3-8, Section C-3 and the Basis for Specification 3.3(2)e(iii). These changes will ensure consistency of reporting requirement references within these Technical Specifications.

Section 5.5.1.6, Plant Review Committee Responsibilities, has been changed by addition of item j. which requires review of all reportable events. This change is consistent with the recommendations of Generic Letter 83-43.

Section 5.5.2.7, Safety Audit and Review Committee (SARC) review responsibilities, has been changed by rewording item g. in accordance with the recommendations of Generic Letter 83-43. The SARC will review all reportable events rather than only those events requiring NRC notification within 24 hours.

Changes to Section 5.6 of the Technical Specifications are proposed to ensure compliance with the reporting requirements of 10 CFR 50.72 and reflect terminology changes consistent with the noted section of 10 CFR 50. The change to Technical Specification 5.6.1.a will commit to notification of the Commission pursuant to 10 CFR 50.72, when applicable. Specification 5.6.1.b is changed to delete reference to 24-hour notifications and for consistency with terminology used elsewhere in the Technical Specifications. A new specification, 5.6.1.c, is added to reference Specification 5.9.2 regarding Licensee Event Reports.

Specifications in Section 5.7 of the Fort Calhoun Station Technical Specifications are being changed to provide specific instructions to be taken in the event of a safety limit violation. The proposed changes will provide for placement of the plant in Hot Shutdown within six (6) hours and notification of the NRC within 24 hours. The other change provides for submittal of a written

report of the circumstances surrounding the safety limit violation within fourteen (14) days to the NRC. These changes are consistent with the Standard Technical Specifications and Fort Calhoun Station Technical Specification 2.01(1).

Specifications in Section 5.9 of these Technical Specifications are being changed to reference 10 CFR 50.73 and to reflect the District's interpretation of the rule. Specification 5.9.2.a has been changed to specifically address the reportability of violations of specifications in sections 1.0 (Safety Limits and Limiting Safety System Settings) and 2.0 (Limiting Conditions for Operation) of these Technical Specifications. This proposed specification is based upon the District's interpretation of item (a)(2)(i)(B) of 10 CFR 50.73.

Item (a)(2)(i)(B) of 10 CFR 50.73 states that the licensee shall report "Any operation or condition prohibited by the plant's Technical Specifications". The District has interpreted this requirement such that the operations and conditions of primary concern are specified in sections 1.0 and 2.0 (titles noted above) of the Fort Calhoun Station Technical Specifications. The District believes that violation of the section 1.0 or 2.0 specifications holds greater potential to adversely impact the health and welfare of the public than violation of the sections 3.0, 4.0, 5.0, or 6.0 specifications (ie., Surveillance Requirements, Design Features, Administrative Controls and Interim Special Technical Specifications, respectively). The District further believes that events holding the greatest potential for adversely affecting the health and welfare of the public should be reported via the LER system.

Based upon the above interpretation, the District proposes a technical specification requiring submittal of an LER for violation of section 1.0 and 2.0 specifications, specifically. The proposed specification 5.9.2.a. is a means of providing clarification of item (a)(2)(i)(B) of 10 CFR 50.73 for District personnel. The District believes this clarification is necessary because there are administrative specifications (eg., fourteen day PRC review and Plant Manager approval of temporary changes to plant procedures) which if violated will not jeopardize plant safety. In the event sections 3.0, 4.0, 5.0, or 6.0 specifications are violated, such violations will be evaluated on a case-by-case basis for adverse impact on plant safety. For the case when violation of a section 3.0, 4.0, 5.0, or 6.0 specification causes violation of a section 1.0 or 2.0 specification, or results in a situation addressed by another reporting requirement of 10 CFR 50.73, an LER will be submitted in accordance with Part 50.73. If the preceding condition is not satisfied, the technical specification violation will be documented in accordance with Specification 5.5.1.6.e.

Section 5.9.1.c, Monthly Operating Report, is changed to read "...Director, Office of Inspection and Enforcement, ..." from "... Office of Management and Program Analysis, ...". This change is consistent with to whom the monthly reports are currently submitted. In addition, the words "to arrive" have been deleted from the statement regarding submittal of the report. The District cannot reasonably assure arrival of a document by any date. This change is also consistent with Standard Technical Specifications.

Section 5.10.1.c has been changed to "Licensee Event Reports (LER)" from "Reportable Occurrence Reports". This change is consistent with the recommendations of Generic Letter 83-43.

Significant Hazards Considerations:

1. This proposed amendment will not involve an increase in the probability or consequence of an accident previously evaluated. The proposed changes concerning Licensee Event and Significant Event reporting are administrative in nature and do not affect the surveillance or operability of any system which functions to prevent or mitigate the consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident than any accident previously evaluated. The proposed changes to the reporting requirements are administrative in nature. The proposed changes do not affect the design or surveillance and operability requirements of the plant systems.

3. The proposed amendment does not involve a reduction in a margin of safety. The proposed changes are administrative in nature and do not affect any design, surveillance, or operability requirements which would have an effect on safety margins.