DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:

1. Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.

2. Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Leakage $> 0.50\ L_a$ shall be to the penetration room.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5 Programs and Manuals (continued)

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. These systems include High Pressure Injection, Low Pressure Injection, Reactor Building Spray, Gaseous Waste Disposal, Makeup and Purification, Chemical Addition and Sampling, and Coolant Treatment. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.4 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents; containment atmosphere samples and airborne iodine concentrations in vital areas under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.5 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in UFSAR Chapter 16, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times 10 CFR Part 20.1001 - 20.2401, Appendix B, Table 2, Column 2;

5.5 Programs and Manuals

5.5.5 Radioactive Effluent Controls Program (continued)

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:

   1. For noble gases; Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   2. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days; less than or equal to a dose rate of 1500 mrems/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

5.5 Programs and Manuals

5.5.5 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

k. Descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the Radiological Effluent Controls of the UFSAR:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. A determination that the change(s) maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations or a determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the station manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire Section 16.11 of the UFSAR as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any changes to Section 16.11 of the UFSAR was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month/year) the change was implemented.

5.5.6 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 5.2.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5 Programs and Manuals (continued)

5.5.7 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.8 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for inspection of each reactor coolant pump flywheel. At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an inplace, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed if the interval measured from the previous such inspection is greater than 6 2/3 years. The interval may be extended up to one year to permit inspections to coincide with a planned outage.

5.5.9 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

5.5 Programs and Manuals

5.5.9 Inservice Testing Program (continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.10 Steam Generator (SG) Tube Surveillance Program

This program provides the controls for SG tube surveillance. The program shall include the following:

a. Examination Methods

Inservice 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.

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

b. Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

For Units 1 and 3, there are a number of steam generator tubes which exceed the tube repair limit as a result of tube end anomalies. These tubes are temporarily exempted from the requirements for sleeving, rerolling or removal from service, until repaired during or before the next Unit 1 and Unit 3 refueling outages (Unit 1 EOC 18, Unit 3 EOC 17 refueling outages, respectively). An analysis has been performed which confirms the operability of Units 1 and 3 will not be impacted with these tubes in service until the next refueling outage on each of these units.

c. Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.10-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.10.d and the inspected tubes shall be verified acceptable per 5.5.10.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

1. The first sample inspection during each inservice inspection of each steam generator shall include:

   a. All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

   b. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

   c. A tube adjacent to any selected tube which does not permit passage of the eddy-current probe for tube inspection.

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

2. Tubes in the following Group(s) may be excluded from the first sample if all tubes in a Group in both OTSG are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

   Group A-1: Tubes within one, two, or three rows of the open inspection lane.

3. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

4. The tubes selected as the second and third samples (if required by Table 5.5.10-1) during each inservice inspection may be subjected to less than a full tube inspection provided:

   a. The tubes selected for these samples include the tubes from those areas of the tubesheet array where tubes with imperfections were previously found.

   b. The inspections include those portions of the tubes where imperfections were previously found.

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

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but no more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

(2) Where special inspections are performed pursuant to 5.5.10.c.2, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection, unless the mechanism of degradation is random in nature.

(3) Where special inspections are performed pursuant to 5.5.10.c.2, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found in the originally rolled region of the rerolled tube, need not be included in determining the Inspection Results Category for the general steam generator inspection.

d. Inspection Intervals

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.10-1 at 40 month intervals fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 months nor more than one fuel cycle after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.10.d.1 and the interval can be extended to a maximum of 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.10-1 during the shutdown subsequent to any of the following conditions:

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

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.

4. After primary to secondary leakage in excess of the limits of Specification 3.4.13, an inspection of the affected steam generator will be performed in accordance with the following criteria:

a. If the leaking tube is in a Group as defined in Section 5.5.10.c.2, all of the tubes in this Group in this steam generator will be inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

b. If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the new roll area in the other steam generator.

c. If the leaking tube is not in a Group as defined in 5.5.10.d.4.a, then an inspection will be performed on the affected steam generator in accordance with Table 5.5.10-1 with an initial inspection sample size of 6% of the tubes in the affected steam generator.

e. Definitions

As used in this specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections.
2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

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

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

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

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

   The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

   The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the Topical Report, BAW-2303, Revision 3.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.10.d.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

TABLE 5.5.10-1 (Page 2 of 2)
STEAM GENERATOR TUBE INSPECTION

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes.
					C-3	Plug or repair defective tubes and perform action for C-3 result of 1st Sample.
			C-3	Plug or repair defective tubes and perform actions for C-3 results on 1st Sample.	N/A	N/A

(continued)

5.5 Programs and Manuals

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
(continued)	C-3	Inspect 6S tubes in the S.G, plug or repair defective tubes and inspect 2S tubes in the other S.G. Perform follow-on inspections in the other S.G. in accordance with results of the above inspection as applied to Table 5.5.10-1 Prompt Notification to NRC pursuant to 10 CFR 50.72	C-1	N/A	N/A	N/A
			C-2	N/A	N/A	N/A
			C-3 (2)	(a) If defects can be localized to an affected area, inspect all tubes in affected area and plug or repair defective tubes. (b) If defects cannot be localized to an affected area, inspect all tubes in this S.G. and plug or repair defective tubes.	C-1 C-2 C-3	N/A N/A N/A

Notes: (1) S=3(N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) Following an 18% random inspection (C-3 category inspection) an unaffected area is identified. The unaffected area will be logically and consistently defined based on generator design, defect location and characteristics. The criteria for accepting an area as unaffected depends on the number of defects found in the sample inspected in that area and are established such that there is a 0.05 or smaller probability of accepting the area as unaffected if it contains 30 or more defective tubes.

5.5 Programs and Manuals (continued)

5.5.11 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.12 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

The VFTP is applicable to the Penetration Room Ventilation System (PRVS), the Control Room Ventilation System (CRVS) Booster Fan Trains, and the Spent Fuel Pool Ventilation System (SFPVS).

a. Demonstrate, for the PRVS, that a dioctyl phthalate (DOP) test of the high efficiency particulate air (HEPA) filters shows $\geq$ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

b. Demonstrate, for the CRVS Booster Fan Trains, that a DOP test of the HEPA filters shows $\geq$ 99.5% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

c. Demonstrate, for the PRVS, that a halogenated hydrocarbon test of the carbon adsorber shows $\geq$ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

5.5 Programs and Manuals

5.5.12 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

e. Demonstrate, for the CRVS Booster Fan Trains, PRVS and SFPVS, that a laboratory test of a sample of the carbon adsorber shows ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

f. Demonstrate, for the PRVS, that the pressure drop across the combined HEPA filters and carbon adsorber banks is < 6 in. of water at the system design flow rate ± 10%.

g. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of water at the system design flow rate ± 10%.

h. Demonstrate, for the SFPVS, that a dioctyl phthalate (DOP) test of the high efficiency particulate air (HEPA) filters shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

i. Demonstrate, for the SFPVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup tanks and the quantity of radioactivity contained in waste gas holdup tanks; and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined. The liquid radwaste quantities shall be determined by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

5.5 Programs and Manuals

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limit for concentration of hydrogen in the waste gas holdup tanks and a surveillance program to ensure the limit is maintained. The limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas holdup tank is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual at the nearest exclusion area boundary, in the event of an uncontrolled release of the tank's contents.

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than 10 curies excluding tritium and dissolved or entrained gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.14 Standby Shutdown Facility (SSF) Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of SSF fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the acceptability of Day Tank and Underground Storage Tank fuel oil for use by determining that the fuel oil viscosity, water and sediment are within limits.

5.5.15 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5 Programs and Manuals

5.5.15 Technical Specifications (TS) Bases Control Program (continued)

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR .

d. Proposed changes that meet the criteria of 5.5.15.b.1 or 5.5.15.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.16 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of safety function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.16 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.17 Backup Method for Determining Subcooling Margin

This program ensures the capability to accurately monitor the Reactor Coolant System Subcooling Margin. The program shall include the following:

a. Training of personnel, and

a. Procedures for monitoring.

5.5.18 KHU Commercial Power Generation Testing Program

The KHU Commercial Power Generation Testing Program shall include the following and shall be met during periods of KHU commercial power generation:

a. Verify upon an actual or simulated actuation signal, each KHU's overhead tie breaker and underground tie breaker actuate to the correct position from an initial condition of commercial power generation every 18 months.

b. Verify upon an actual or simulated actuation signal, each KHU's frequency is $\leq$ 66 Hz in $\leq$ 23 seconds from an initial condition of commercial power generation every 18 months.

5.5 Programs and Manuals

5.5.18 KHU Commercial Power Generation Testing Program (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the KHU Commercial Power Generation Testing Program surveillance frequencies.

5.5.19 Lee Combustion Turbine Testing Program

The Lee Combustion Turbine (LCT) Testing program shall include the following and shall be met when a LCT is used to comply with Required Actions of Specification 3.8.1, "AC Sources-Operating" or as a emergency power source as allowed by LCO 3.8.2, "AC Sources-Shutdown":

a. Verify an LCT can energize both standby buses using 100kV line electrically separated from system grid and offsite loads every 12 months.

b. Verify an LCT can supply equivalent of one Unit's maximum safeguard loads plus two Unit's MODE 3 loads when connected to system grid every 12 months.

c. Verify an LCT can provide equivalent of one Unit's maximum safeguard loads within one hour through 100kV line electrically separated from system grid and offsite loads every 18 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Lee Combustion Turbine Testing Program surveillance frequencies.

5.5.20 Battery Discharge Testing Program

The Battery Discharge Testing Program shall include the following and shall be met for batteries used to comply with LCO 3.8.3, "DC Sources Operating."

a. Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months. This frequency shall be reduced to 12 months when battery shows degradation, or has reached 90% of the expected life with capacity < 100% of manufacturer's rating, and 24 months when battery has reached 90% of the expected life with capacity ≥ 100% of manufacturer's rating.

5.5 Programs and Manuals

5.5.20 Battery Discharge Testing Program (continued)

b. If battery capacity is determined to be < 80% of the manufacturer's rating an OPERABILITY evaluation shall be initiated immediately and completed within the guidelines of the Oconee OPERABILITY program. If the OPERABILITY evaluation determines the battery OPERABLE, battery capacity shall be restored to ≥ 80% of the manufacturer's rating within a time frame commensurate with the safety significance of the issue. Otherwise, the battery shall be declared inoperable and the applicable Condition of Specification 3.8.3 shall be entered.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Battery Discharge Testing Program surveillance frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year.

The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

Core operating limits shall be established, determined and issued in accordance with the following:

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin limit for Specification 3.1.1;

2. Moderator Temperature Coefficient limit for Specification 3.1.3;

3. Physical Position, Sequence and Overlap limits for Specification 3.2.1 Rod Insertion Limits;

4. AXIAL POWER IMBALANCE operating limits for Specification 3.2.2;

5. QUADRANT POWER TILT (QPT) limits for Specification 3.2.3;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985;

(2) NFS-1001A, Reload Design Methodology, April 1984;

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989;

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992;

(5) BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989;

(6) BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February, 1994;

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, August, 1994;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, February, 1995 and

(9) DPC-NE-3005-P, UFSAR Chapter 15 Transient Analysis Methodology, DPC, JUL97.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

5.6.8 Steam Generator Tube Inspection Report (continued)

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged or repaired.

4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC shall be reported prior to resumption of plant operation. The written report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

TECHNICAL SPECIFICATIONS

MARKED UP PAGES

FOR INFORMATION ONLY

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing.

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

INSERT FROM FOLLOWING PAGE

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Leakage $> 0.50\ L_a$ shall be to the penetration room.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. These systems include High Pressure Injection, Low Pressure Injection, Reactor Building Spray, Gaseous Waste Disposal, Makeup and Purification, Chemical Addition and Sampling, and Coolant Treatment. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

SUBSEQUENT PAGES REPAGINATED

Insert to TS 5.5.2 on Page 5.0-8

Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:

1. Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.

2. Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

5.5 Programs and Manuals

5.5.5 Radioactive Effluent Controls Program (continued)

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

2. A determination that the change(s) maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations or a determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the station manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire Section 16.11 of the UFSAR as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any changes to Section 16.11 of the UFSAR was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month/year) the change was implemented.

5.5.6 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 5.2.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.7 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~Regulatory Guide 1.35, Revision 3, 1990~~.

INSERT FROM FOLLOWING PAGE

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

Insert to TS 5.5.7 on Page 5.0-11

Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

ATTACHMENT 3

TECHNICAL JUSTIFICATION

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ATTACHMENT 3

TECHNICAL JUSTIFICATION

1.0 Description of Proposed Changes

This proposed amendment revises Technical Specification (TS) 5.5.2, "Containment Leakage Rate Testing Program" and TS 5.5.7, "Pre-Stressed Concrete Containment Tendon Surveillance Program" as follows:

The proposed revision to TS 5.5.2 requests two clarifications to the requirements of Regulatory Guide (RG) 1.163 (Reference 1) that would:

- Permit visual examination of containment concrete surfaces and the tendon system up to 90-days prior to refueling outages in which a 10 CFR 50, Appendix J Containment Type A test will be performed. The present requirements do not permit these visual exams prior to beginning the refueling outage in which a Type A test is performed.

- Clarify that general visual examinations of containment pressure retaining metallic surfaces are required three times every 10 years, and that only those general visual examinations performed in conjunction with each Type A test need be performed during shutdown.

The proposed TS 5.5.7 change replaces the current RG 1.35 (Reference 3) containment tendon test program requirements with those of ASME XI, IWL. This change is submitted per the requirements of 10CFR50.55a(g)(5)(ii) to conform the TSs to the revised program specified by 10 CFR 50.55a(g)(4). The proposed change applies to tendon testing performed after January 1, 2000. The next tendon testing which must use the revised test program will be during the Unit 3 end-of-cycle (EOC) 18 refueling outage, which is currently scheduled to begin April 17, 2000. The testing frequencies will be per IWL (i.e., unchanged from the 5-year interval of the current program). Surveillance Requirement (SR) 3.0.2 will no longer apply since IWL requires the inspection to be performed every five years, +/- 12 months.

Approval of these changes is requested by February 17, 2000 in order to provide sufficient time to implement the change and perform required testing prior to the beginning of the next Unit 3 refueling outage.

2.0 Background

On August 8, 1996, the NRC published in the Federal Register (61 FR 41303) changes to 10 CFR 50.55a that invoked the

requirements of the ASME XI, Subsections IWE and IWL. This rule change required that utilities perform inservice inspections of containments in accordance with the 1992 Edition of the ASME Code, through 1992 Addenda (with specific exceptions and limitations). For Oconee, this required that surveillance tests be performed in a manner similar to that specified in RG 1.35, as approved by the NRC in the current Improved TSs. In addition, the rule change allows utilities to take credit for Reactor Building Post-Tensioning System tests to satisfy the IWL provisions for a 5-year period from the effective date of the rule change for tests that were performed in accordance with RG 1.35 before the effective date of the rule change. However, at the time the current TSs were approved, Oconee had not yet performed tests in accordance with RG 1.35 and was not prepared to perform the tests in accordance with the ASME Code. Therefore, the current TSs satisfied the Expedited Examination of Containment Post-Tensioning System requirements contained in the rule change on an interim basis until final implementation of the rule.

Duke Energy, Inc. (Duke) has performed the aforementioned post-tensioning system tests per RG 1.35 during Unit 1 RFO EOC17. These tests will also be performed per RG 1.35 during the next Unit 2 RFO scheduled for late 1999. Duke will be prepared to perform the tests in accordance with the ASME Code beginning with the next Unit 3 refueling outage. The TS changes described in the following Sections 3.2 and 3.3 of this Technical Justification fulfill the foregoing requirement.

3.0 Technical Justification

3.1 Changes to Technical Specification 5.5.2

TS Administrative Controls, Section 5.0, 5.5.2 "Containment Leakage Rate Testing Program" requires, in part:

> "This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A."

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TECHNICAL JUSTIFICATION

Duke is proposing to amend this requirement to add the following text:

> "*Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:*
>
> 1. *Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.*
>
> 2. *Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.*"

This change is proposed for the following reasons:

Regulatory Position C.3 of RG 1.163 requires, in part, visual examination of accessible concrete surfaces and post-tensioning system component surfaces to be performed during refueling outages. The requirement to perform the examinations during a refueling outage may unnecessarily increase the duration of refueling outages to complete these examinations along with other required inspections and testing. Performing all, or part, of the visual examinations prior to the beginning of a refueling outage eliminates the possibility that these examinations could affect the outage critical path.

The purpose of the containment visual examinations required by 10 CFR 50, Appendix J and RG 1.163, Regulatory Position C.3 is to detect deterioration that could affect the containment leak-tightness or structural integrity. Whether these examinations are performed during operation or when shutdown has no

impact on the quality of the inspection, provided all accessible interior and exterior surfaces are examined. Therefore, Duke is requesting approval to perform all, or portions, of these visual examinations on accessible interior and exterior surfaces during either plant operation or shutdown, with the exception (as stated above) that examinations of the metallic pressure retaining components of the containment system performed prior to each Type A test shall be performed during the outage in which the Type A test is performed.

RG 1.163, Regulatory Position C.3 requires, in part, that visual examinations of accessible concrete and post-tensioning system surfaces be performed prior to each Type A test. The RG also requires additional examinations of these component surfaces to be conducted during two other refueling outages between Type A tests if the interval for the Type A test has been extended to 10 years. Oconee currently performs Type A tests at 10-year intervals.

ASME XI, Subsection IWL concrete and post-tensioning system examinations are performed at five-year intervals.

The effect of the proposed change is as follows:

1) Duke will perform a general visual examination of accessible concrete and post-tensioning system surfaces during shutdown for each Type A test in accordance with 10CFR50, Appendix J and RG 1.163.

2) Duke will perform examinations of accessible concrete and post-tensioning system surfaces every five years in accordance with the ASME Code, Section XI, Subsection IWL. These examinations need not be performed during plant outages.

3) If an IWL examination is completed just prior to a scheduled Type A test, the IWL examination may be credited towards satisfying the general visual examination of concrete and post-tensioning system surfaces required by RG 1.163 so that a total of two examinations need only be performed during any 10 year interval. When this occurs, the IWL examinations may be performed as much as 90 days prior to the start of the refueling outage in which the Type A test is scheduled.

A five-year examination frequency is generally considered adequate for passive structural components, and other

nuclear safety related concrete structures that are typically examined at five-year frequencies in order to satisfy requirements of the Maintenance Rule. Furthermore, if IWL examinations are scheduled during an outage when a Type A test is also scheduled, sufficient time may not be available during the refueling outage to perform these examinations and the Type A test without unnecessarily extending the length of the refueling outage. Specifying that concrete and post-tensioning system visual examinations required by 10 CFR 50, Appendix J be performed within 90 days of the start of a refueling outage in which a Type A test is scheduled is sufficient to detect evidence of deterioration which may affect the structural integrity of the containment, and satisfy the intent of 10 CFR 50, Appendix J that these examinations be performed immediately prior to the Type A test.

When possible, Duke intends to perform these general visual examinations concurrently with general visual examinations required by ASME XI, IWE, Table IWE-2500-1, Examination Category E-A, Item E1.11 during each ISI Period, as required by 10 CFR 50.55a (b)(2)(x)(E). The proposed changes will facilitate scheduling these examinations such that they can be credited towards satisfying both 10 CFR 50, Appendix J and ASME XI, IWE general visual examination requirements. The ASME Code does not require that general visual examinations be performed during refueling outages. Because these visual examinations will be performed during each ISI Period, a minimum of three examinations shall be performed every ten years. Therefore, the revised TSs will provide a level of quality and safety equivalent to the current TS requirements.

The proposed provision to evaluate pre-outage examination results in the unlikely event of an occurrence capable of affecting the integrity of the containment system provides assurance that requirements of TS 5.5.2 remain satisfied.

3.2 Changes to Technical Specification 5.5.7

TS Administrative Controls, Section 5.0, 5.5.7 "Pre-stressed Concrete Containment Tendon Surveillance Program" states:

> "This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and

acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1990.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies."

Duke is proposing to amend this requirement to read as follows:

"This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with *Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).*

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies."

The NRC amended 10 CFR 50.55a to require utilities to implement ASME XI, IWL and to complete expedited examinations no later than September 9, 2001. After completing the expedited post-tensioning system examinations during Unit 2 RFO EOC17, all future inservice inspections of containment post-tensioning systems at Oconee will comply with ASME XI, IWL. This makes it necessary to amend this technical specification to reference ASME XI and delete the reference to RG 1.35. Since the tendon inspection frequencies will be in accordance with ASME XI, IWL, the provisions of SR 3.0.2 will no longer apply and are deleted from TS 5.5.7.

3.3 Changes to SLC 16.6.2, Containment Tendon Surveillance Program

The current containment surveillance program was originally included in custom TS sections (CTS) 3.6.7 and 4.4.2. Additionally, the prescribed lower limit (PLL) and the minimum required value (MRV) of tendon force for each group of tendons were provided in SLC 16.6.2 to meet the commitment described in Section 2.0 above.

ATTACHMENT 3

TECHNICAL JUSTIFICATION

During the conversion from CTS to Improved TSs, the details of the containment tendon surveillance program required by ITS 5.5.7 were relocated from CTS 3.6.7 and 4.4.2 to SLC 16.6.2.

In the time since this Improved TS has been approved, an Inservice Inspection (ISI) Program that is in accordance with ASME XI, IWL has been implemented at Oconee as required by 10 CFR 50, Section 55a(g)(4). This ISI Program includes concrete pressure retaining components and post-tensioning systems of concrete containments. The details for tendon surveillance as presently included in SLC 16.6.2 are in accordance with RG 1.35. After Unit 2 RFO EOC17, all tendon surveillances at Oconee will be performed in accordance with ASME XI, IWL. The details for the inservice inspection of containment post-tensioning system components are included in the Containment Inservice Inspection Plan (Reference 4). Revising SLC 16.6.2 to reflect the requirements of ASME XI, IWL would be an unnecessary duplication of requirements specified in the Containment ISI Plan.

On approval of the above change to TS 5.5.7, Duke will revise SLC 16.6.2 as indicated in Enclosures A and B to this Attachment. These changes will reference the ASME XI Containment ISI Program and remove these details from the SLC for inservice inspection of the concrete containment post-tensioning system. Figures 16.6.2-1, 16.6.2-2, and 16.6.2-3 (PLLs and MRVs for the Dome, Hoop, and Vertical Tendons respectively) will continue to be maintained in this SLC.

4.0 References

1. Regulatory Guide 1.163, "Performance-Based Containment Leak-test Program," September 1995.

2. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1992 Edition, with 1992 Addenda.

3. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons In Prestressed Concrete Containments," Revision 3, July 1990.

4. Duke Document No. O-62-CISI-0001, "First Interval Containment Inservice Inspection Plan, Oconee Nuclear Station Units 1, 2, and 3."

ATTACHMENT 3

TECHNICAL JUSTIFICATION

Enclosure 1

Revised SLC 16.6.2 Pages

Remove Pages	Insert Pages
16.6.2-3	16.6.2-3
16.6.2-4	16.6.2-4
16.6.2-5	16.6.2-5
16.6.2-6	16.6.2-6
16.6.2-7	16.6.2-7
16.6.2-8	----
16.6.2-9	----
16.6.2-10	----
16.6.2-11	----
16.6.2-12	----

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.2.1	---------------------------NOTE------------------------- This SR may be conducted during MODE 1 provided design conditions regarding loss of adjacent tendons are satisfied at all times. --- Perform inservice examinations of concrete containment post-tensioning systems in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).	As specified by IWL and 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

Figure 16.6.2-1
Dome Tendon PLLs and MRVs

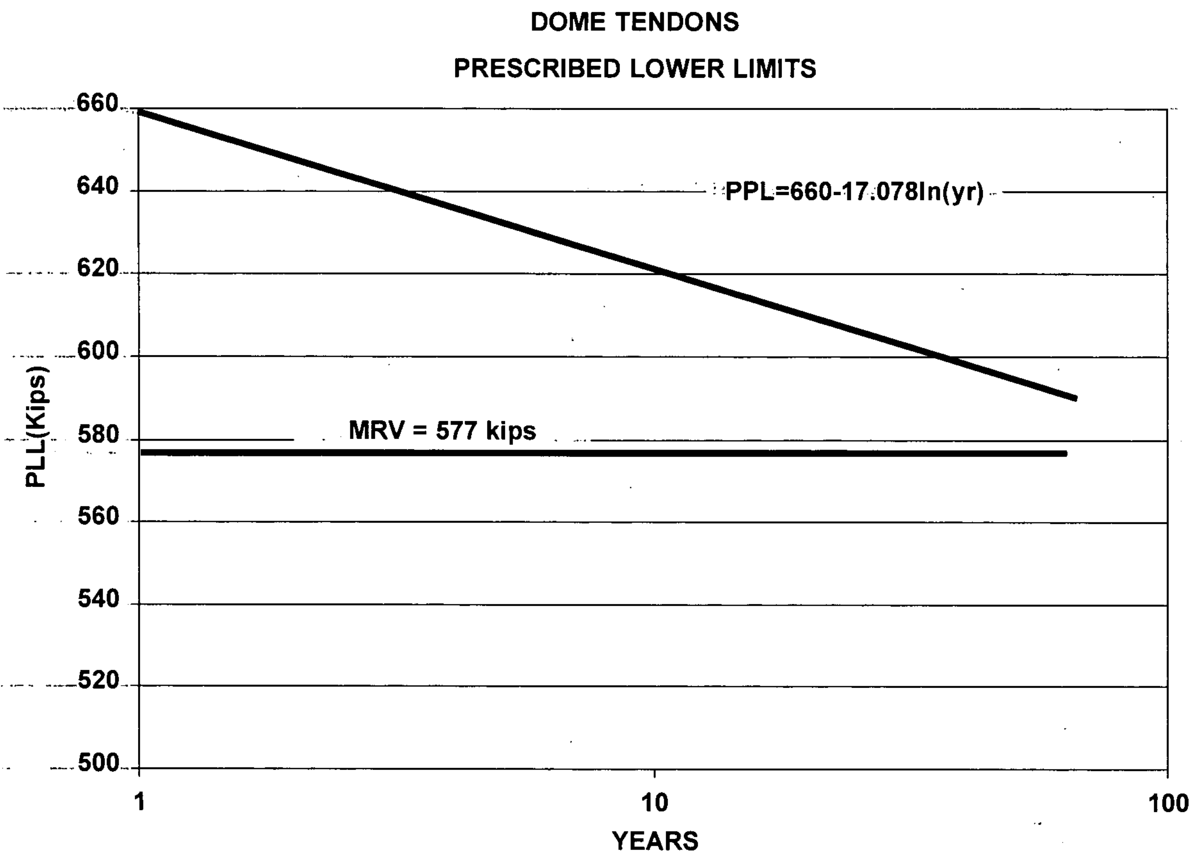


Figure 16.6.2-2
Hoop Tendon PLLs and MRVs

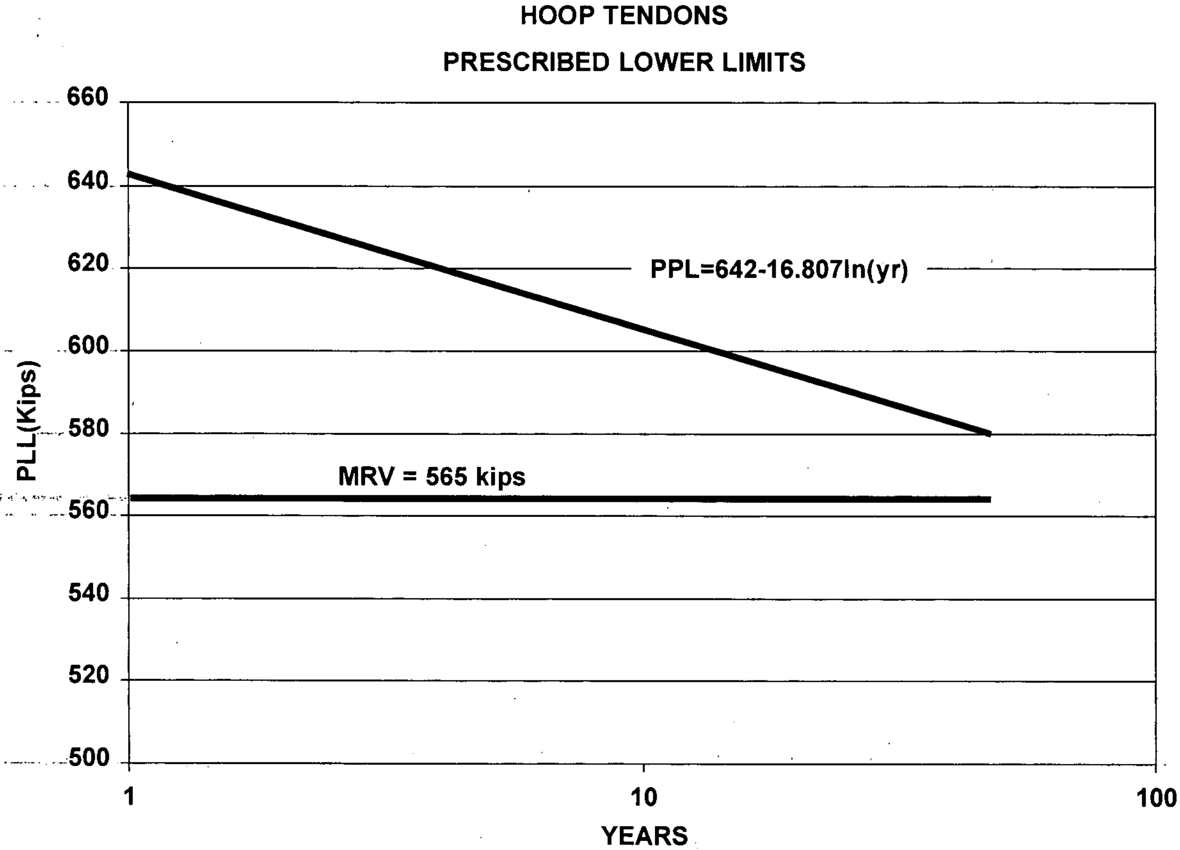
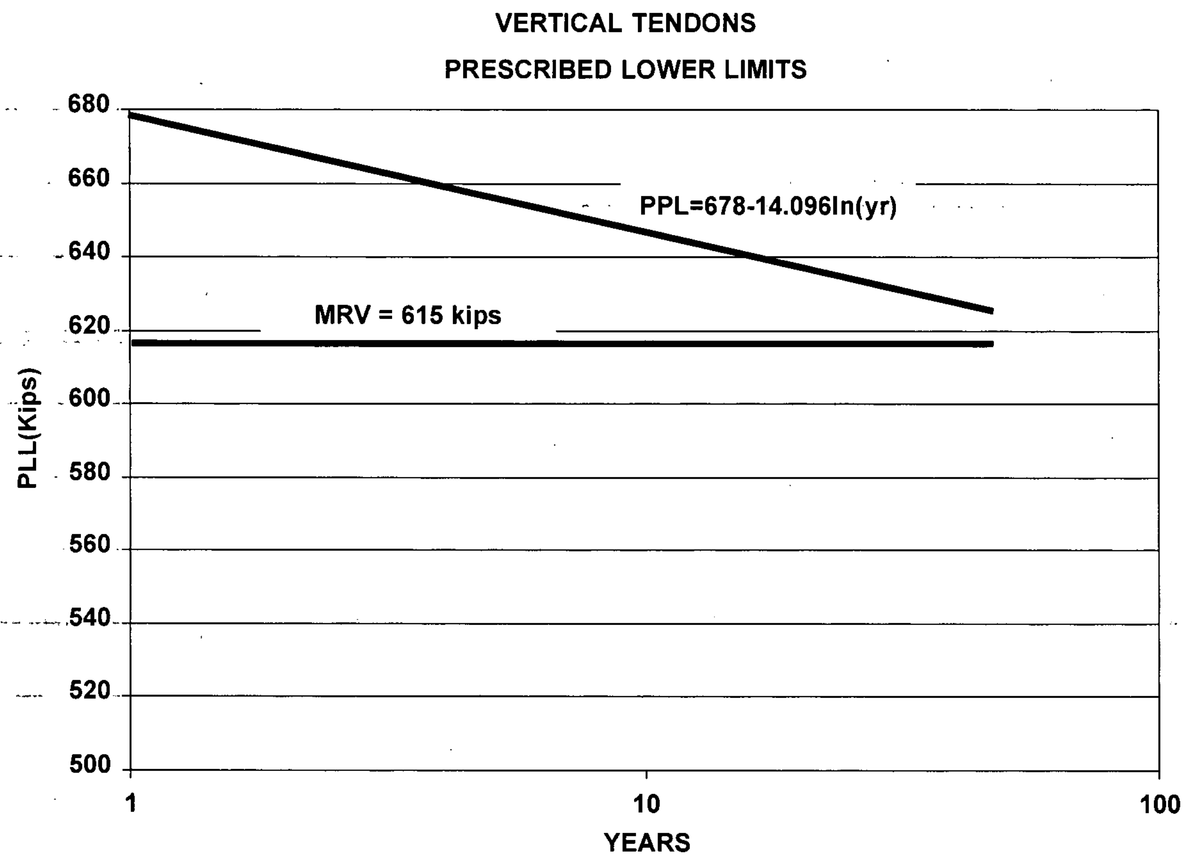


Figure 16.6.2-3
Vertical Tendon PLLs and MRVs



BASES

The NRC amended 10 CFR 50.55a on August 8, 1996 to include examination of Class MC and CC components. This rule became effective September 9, 1996, and incorporates by reference Subsections IWE and IWL of the ASME Code, Section XI. Subsections IWE and IWL provide requirements for inservice inspection, repair/replacement, and testing of both steel and concrete containments, including post-tensioning systems. Oconee has implemented a Containment Inservice Inspection Plan (Reference 1) to comply with the requirements of 10 CFR 50.55a. Prior to the implementation of the Containment Inservice Inspection Plan, containment structural integrity was verified in accordance with Regulatory Guide 1.35, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments, and the details were included in this SLC. Because the Oconee Containment Inservice Inspection Plan now contains the necessary details for inservice inspection of the containment, the details previously included in this SLC have been removed. The required actions and completion times upon detection of conditions of abnormal degradation have not been changed. Additionally, the Minimum Required Values (MRVs) and Prescribed Lower Limits (PLLs) for the dome, hoop, and vertical tendons have not been revised, and will continue to be maintained in this SLC. The surveillance requirements have been condensed to the following:

SR 16.6.2.1

This SR performs inservice examinations of post-tensioning systems of concrete containments in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 55a. The details of this SR are outlined in the Oconee Containment Inservice Inspection Plan.

Any condition which fails to meet the acceptance standards of IWL-3000 shall be considered as an indication of abnormal degradation of the reactor building. Additionally, any unacceptable condition listed in 10 CFR 50.55a(b)(2)(ix) shall also be considered as an indication of abnormal degradation of the reactor building.

References

1. Duke Document No. O-62-CISI-0001, "First Interval Containment Inservice Inspection Plan, Oconee Nuclear Station Units 1, 2, and 3."

ATTACHMENT 3

TECHNICAL JUSTIFICATION

Enclosure 2

Current SLC 16.6.2 with Marked Pages

16.6 ENGINEERED SAFETY FEATURES

FOR INFORMATION ONLY

16.6.2 Containment Tendon Surveillance Program

COMMITMENT The structural integrity of the containment shall be maintained.

The Reactor Building Post-Tensioning System shall meet the minimum required values (MRVs) and Prescribed Lower Limits (PLLs) as specified in this Selected Licensee Commitment. The required MRVs and PLLs which shall be used as limits during the conduct of the SRs are specified in Figures 16.6.2-1, 16.6.2-2 and 16.6.2-3.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Abnormal degradation of containment structural integrity indicated by average of all measured prestressing forces for any group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group.	A.1 Restore containment to required level of structural integrity	72 hours
	<u>OR</u>	
	A.2.1 Verify that containment structural integrity is maintained, by performing an engineering evaluation of the containment structural integrity.	72 hours
	<u>AND</u>	
	A.2.2 Submit Tendon Surveillance Report in accordance with ITS 5.6.7.	30 days

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CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Abnormal degradation of the containment structural integrity other than Condition A.	B.1	Restore containment to required level of structural integrity	15 days
	OR		
	B.2.1	Performing an engineering evaluation to verify that containment structural integrity is maintained.	15 days
	AND		
	B.2.2	Submit Tendon Surveillance Report in accordance with ITS 5.6.7.	30 days
C. Required Action and associated Completion Time of Required Action A.1, A.2.1, B.1 or B.2.1 not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.2.1	-----NOTE----- 1. This SR may be conducted during MODE 1 provided design conditions regarding loss of adjacent tendons are satisfied at all times. 2. ~~Frequency may be modified in accordance with ASME Section XI, Subsection IWL.~~ ---------- INSERT 1 ON FOLLOWING PAGE ~~Determine that a random, but representative, sample of at least eleven tendons (five hoop, three vertical, three dome) each have an observed lift-off force within the predicted limits established for each tendon group.~~	INSERT 2 ON FOLLOWING PAGE ~~5 years~~
SR 16.6.2.2	~~Verify on a tendon from each group that tendon wires are free of corrosion, cracks, and damage, and minimum tensile strength of 240,000 psi (guaranteed ultimate tensile strength of the wire material) exists for at least three wire samples (one from each end and one at mid-length) cut from the removed wire.~~	~~5 years~~
SR 16.6.2.3	~~Verify for tendons detensioned for inspection that SR retensions to a force at least equal the force recorded prior to detensioning or the predicted value at the time of inspection, whichever is greater, but do not exceed 70% of the guaranteed ultimate tensile strength of the tendon wire material.~~	~~5 years.~~

Insert to SLC 16.6.2 on Page 16.6.2-3

Insert 1

Perform inservice examinations of concrete containment post-tensioning systems in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

Insert 2

As specified by IWL and 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

SURVEILLANCE		FREQUENCY
SR 16.6.2.4	Verify acceptability of the sheathing filler grease by assuring that: 1. No free water is present and no changes in the presence or physical appearance of the sheathing filler grease occur. 2. Amount of grease replaced does not exceed 5% of the net duct volume when injected at +/-10% of the specified installation pressure. 3. Minimum grease coverage exists for the different parts of the anchorage system. 4. Reactor building exterior surface does not exhibit grease leakage that could affect reactor building integrity. 5. Chemical properties of the sheathing filler grease are within the following tolerance limits: Water Content: 0 - 10% (dry wt.) Chlorides: 0 - 10 ppm Nitrates: 0 - 10 ppm Sulfides: 0 - 10 ppm Reserve Alkalinity (Base Numbers): > 50% of installed value; > 0 (for older grease)	5 years
SR 16.6.2.5	Verify no abnormal degradation exists by visual examination of tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection.	5 years

SURVEILLANCE		FREQUENCY
~~SR 16.6.2.6~~	~~-----NOTE----- This inspection may be performed prior to the Type A containment leakage rate test. ---------- The exterior surface of the reactor building(s) should be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage.~~	~~5 years~~

Figure 16.6.2-1
Dome Tendon PLLs and MRVs

FOR INFORMATION ONLY

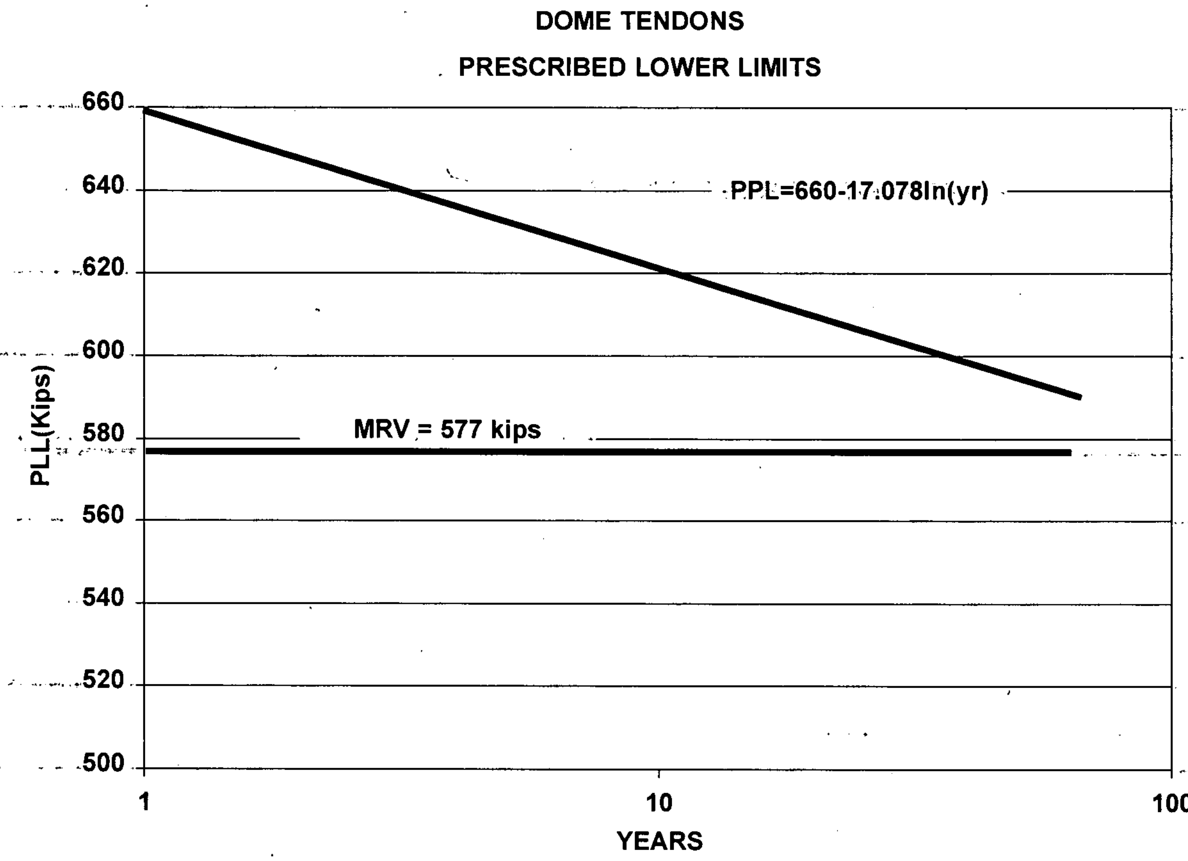


Figure 16.6.2-2
Hoop Tendon PLLs and MRVs

FOR INFORMATION ONLY

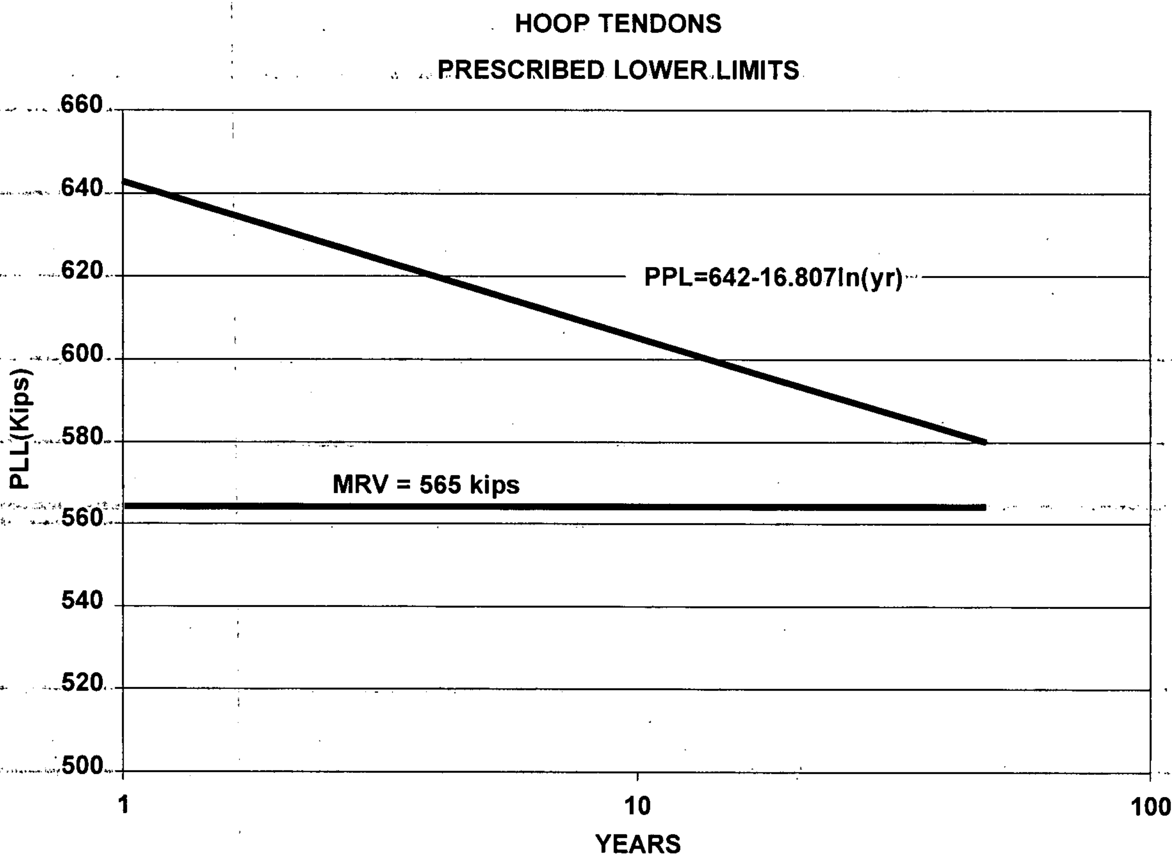
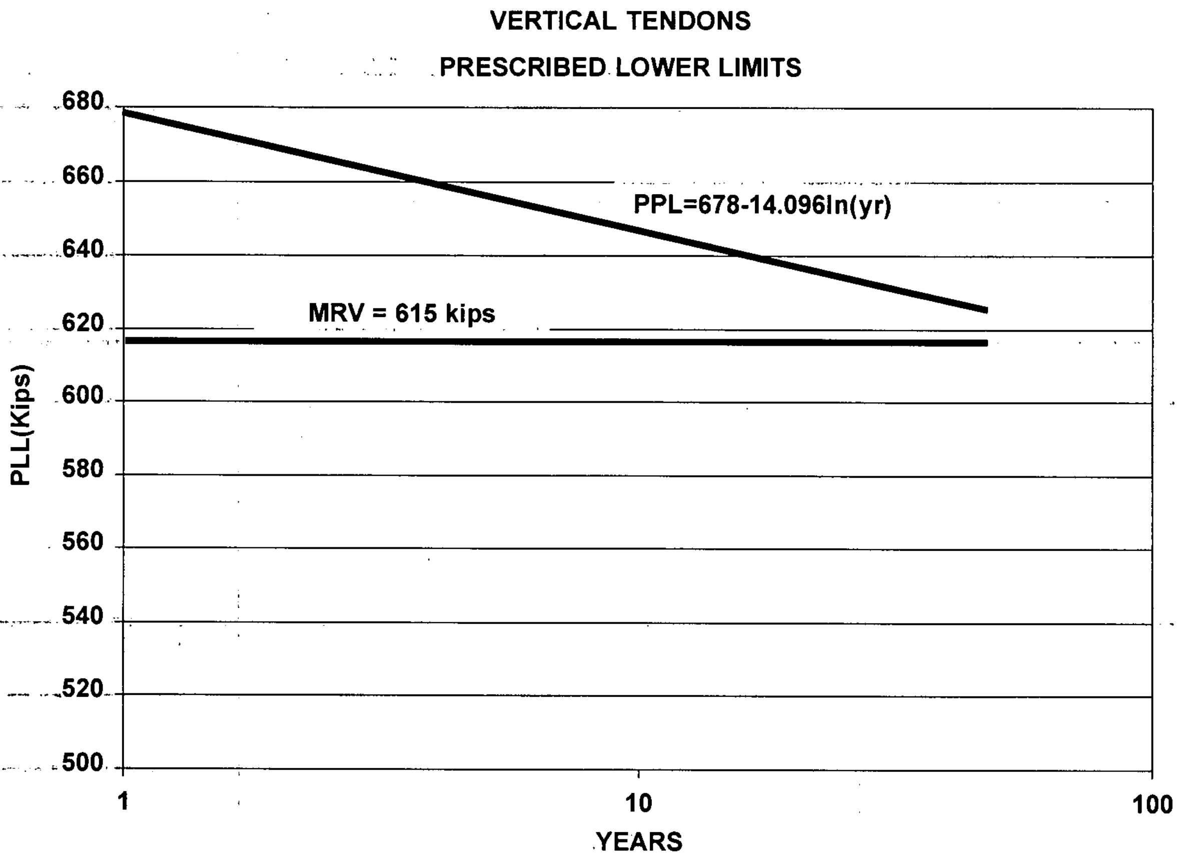


Figure 16.6.2-3
Vertical Tendon PLLs and MRVs

FOR INFORMATION ONLY



INSERT FROM FOLLOWING PAGE

BASES

Some requirements in this SLC were relocated from CTS 3.6.7 and 4.4.2 during the conversion to ITS. The Selected Licensee Commitment (SLC) provides the details of the Containment Tendon Surveillance Program required by ITS 5.5.7. This SLC prescribes Minimum Required Values (MRV's) and Prescribed Lower Limits (PLL's). In a letter dated July 2, 1997, Duke committed to provide a SLC which prescribes MRVs and PLLs in support of the Reactor Building Post-Tensioning (RBPT) System surveillances which are performed in accordance with ITS SR 3.6.1.3 and ITS 5.5.7, Pre-Stressed Concrete Containment Tendon Surveillance Program. In letters dated October 30, 1996, and April 22, 1997, Duke requested a Technical Specification amendment to convert from a Reactor Building Post Tensioning (RBPT) System surveillance methodology of testing pre-designated tendons to a more industry-wide methodology as prescribed in Regulatory Guide 1.35 Revision 3. Regulatory Guide 1.35 Revision 3 requires testing of tendons which are randomly selected from the population of in-service tendons.

Acceptance criteria are given in terms of PLL's and MRV's. The required MRVs and PLLs which shall be used as limits during the conduct of the surveillances are provided in Figures 16.6.2-1, 16.6.2-2, and 16.6.2-3. These figures contain the dome, hoop, and vertical tendon MRVs and PLLs, respectively, for all three units.

Provisions have been made for an inservice inspection program intended to provide sufficient evidence that the integrity of the Reactor Building is being preserved. This program will be conducted in accordance with the guidance of Regulatory Position C of Regulatory Guide 1.35, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments, Revision 3 dated July 1990. Regulatory Guide 1.35 describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for ungrouted tendons in prestressed concrete reactor buildings of light-water-cooled reactors. The inservice inspection program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete reactor buildings throughout the life of the plant.

Prior to implementation of Regulatory Guide 1.35 methodology in accordance with this specification, RBPT System surveillances were performed by examining specific, pre-designated test tendons. Therefore, this specification conservatively identifies the date of the last surveillance performed for each unit under the superseded CTS 4.4.2, and measures the periodicity of future inspections from these dates.

Seating forces for all tendons were documented at the time of installation, thus providing one data point. A second point will be obtained from data obtained during the initial tendon surveillance for each unit. The data from the initial surveillance is considered reliable since any error due to tensioning and retensioning had not been introduced. This data will be averaged on a per unit basis and used in the trend analysis along with new data obtained from the new proposed surveillance program in accordance with Regulatory Guide 1.35.

Insert to SLC 16.6.2 on Page 16.6.2-9

The NRC amended 10 CFR 50.55a on August 8, 1996 to include examination of Class MC and CC components. This rule became effective September 9, 1996, and incorporates by reference Subsections IWE and IWL of the ASME Code, Section XI. Subsections IWE and IWL, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3), provide requirements for inservice inspection, repair/replacement, and testing of both steel and concrete containments, including post-tensioning systems. Oconee has implemented a Containment Inservice Inspection Plan (Reference 1) to comply with the requirements of 10 CFR 50.55a. Prior to the implementation of the Containment Inservice Inspection Plan, containment structural integrity was verified in accordance with Regulatory Guide 1.35, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments, and the details were included in this SLC. Because the Oconee Containment Inservice Inspection Plan now contains the necessary details for inservice inspection of the containment, the details previously included in this SLC have been removed. The required actions and completion times upon detection of conditions of abnormal degradation have not been changed. Additionally, the Minimum Required Values (MRVs) and Prescribed Lower Limits (PLLs) for the dome, hoop, and vertical tendons have not been revised, and will continue to be maintained in this SLC. The surveillance requirements have been condensed to the following:

SR 16.6.2.1

This SR performs inservice examinations of post-tensioning systems of concrete containments in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 55a. The details of this SR are outlined in the Oconee Containment Inservice Inspection Plan.

Any condition which fails to meet the acceptance standards of IWL-3000 shall be considered as an indication of abnormal degradation of the reactor building. Additionally, any unacceptable condition listed in 10 CFR 50.55a(b)(2)(ix) shall also be considered as an indication of abnormal degradation of the reactor building.

References

1. Duke Document No. O-62-CISI-0001, "First Interval Containment Inservice Inspection Plan, Oconee Nuclear Station Units 1, 2, and 3."

SR 16.6.2.1

This SR determines that a random, but representative, sample of at least eleven tendons (five hoop, three vertical, three dome) each have an observed lift-off force within the predicted limits established for each tendon group. For each subsequent inspection, one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

1. If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability.

2. If the measured prestressing force of the selected tendon in a group lies between 95% of the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for the tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing forces of any two adjoining tendons fall below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence. The conditions shall be considered as an indication of abnormal degradation of the reactor building(s).

3. If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be fully investigated and additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the reactor building.

4. If the average of all measured prestressing forces for any group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as abnormal degradation of the reactor building.

5. If the measured prestressing forces from consecutive surveillances for the same tendon, or tendons in a group, indicate a trend of prestress loss larger than expected and the resulting prestressing forces are likely to be less than the minimum required for the group before the next scheduled surveillance, additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the reactor building.

SR 16.6.2.2

This SR performs tendon detensioning, inspections, and material tests on a tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify any broken or damaged wires and to determine the following conditions over the entire length of a removed tendon wire sample (this wire sample should be the broken wire if so identified):

1. Tendon wires are free of corrosion, cracks, and damage, and
2. Minimum tensile strength of 240,000 psi (guaranteed ultimate tensile strength of the wire material) exists for at least three wire samples (one from each end and one at mid-length) cut from the removed wire.

Failure to meet these requirements shall be considered as an indication of abnormal degradation of the reactor building.

SR 16.6.2.3

This SR retensions tendons detensioned for inspection to a force at least equal the force recorded prior to detensioning or the predicted value at the time of inspection, whichever is greater, but do not exceed 70% of the guaranteed ultimate tensile strength of the tendon wire material. Tendon seating force tolerance shall be -0 / +6%. During retensioning of these tendons, change in load versus elongation should be measured at varying levels of force. The following table provides levels of force, pressure, and elongation at which measurements should be taken:

	Force (Kips)	Pressure (psi)	Elongation (In)
PTF			
Step 1			
Step 2			
LOF			
OSF			

Where:
Total Elongation (actual) = (LOF-PTF) Elongation
PTF - Pretensioning Force necessary to bring the tendon into a slightly stressed condition to remove slack and seat the buttonheads.
Step 1-2 - An intermediate force approximately equally spaced between PTF and LOF.
LOF - Lock Off Force at which the tendon is seated on the shims.
OSF = Overstress Force at which the maximum elongation is measured.

If the elongation corresponding to a specific load differs by more than 10% from that recorded during the original installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires at anchorages. This condition shall be considered as an indication of abnormal degradation of the reactor building.

SR 16.6.2.4

This SR verifies acceptability of the sheathing filler grease by assuring that:

1. No free water is present and no changes in the presence or physical appearance of the sheathing filler grease occur.
2. Amount of grease replaced does not exceed 5% of the net duct volume when injected at +/-10% of the specified installation pressure.
3. Minimum grease coverage exists for the different parts of the anchorage system.
4. Reactor building exterior surface does not exhibit grease leakage that could affect reactor building integrity.
5. Chemical properties of the sheathing filler grease are within the specified tolerance limits.

Failure to meet these requirements shall be considered an indication of potential abnormal degradation of the reactor building.

SR 16.6.2.5

As an assurance of the structural integrity of the reactor building(s), tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. Tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load bearing components of the anchorages. Top and bottom grease caps of all vertical tendons shall be visually inspected to detect grease leakage or grease cap deformations. The surrounding concrete should also be checked visually for indication of any abnormal condition.

Significant grease leakage, grease cap deformation or abnormal concrete condition shall be considered as an indication of abnormal degradation of the reactor building.

SR 16.6.2.6

The exterior surface of the reactor building(s) should be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage. Each of these conditions can be considered as evidence of abnormal degradation of structural integrity of the reactor building(s). This inspection may be performed prior to the Type A containment leakage rate test.

REFERENCES

1. Duke letter to NRC dated 10/30/97
2. Duke letter to NRC dated 4/22/97
3. Duke letter to NRC dated 7/2/97
4. Duke letter to NRC dated 9/3/97
5. Duke letter to NRC dated 9/4/97

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Duke Energy Corporation (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10CFR50.92. This ensures operation of the facility in accordance with the proposed amendment will not:

A. Involve a significant increase in the probability or consequences of an accident previously evaluated?

NO.

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no significant effect on accident probabilities or consequences. The containment is not an accident initiating system or structure; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The containment serves an important function to mitigate consequences of postulated accidents previously evaluated and the examination frequencies proposed in this amendment will not result in a reduction in the capacity of the containment to meet its intended function. The requested flexibility in scheduling containment visual examinations has no significant impact on the validity of the examinations or of containment structural integrity.

Additionally, the change to Technical Specification 5.5.7 and the planned revision to Selected Licensee Commitment 16.6.2 described in this amendment application reflect the adoption of an ASME Section XI, Subsection IWE and IWL Inservice Inspection Program as required by 10 CFR 50 Section 55a(g)(4). Implementation of this program will not result in a reduction in the capacity of the containment to meet its intended function.

Therefore, the probability or consequences of an accident previously evaluated will not be increased by approval of the requested changes.

B. Create the possibility of a new or different kind of accident from the accident previously evaluated?

NO.

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant that would introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators, since the containment functions primarily as an accident mitigator.

C. Involve a significant reduction in a margin of safety?

NO.

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation, including the performance of the containment. This component is already capable of performing as intended, and its function is verified by visual examination, post-tensioning system examinations, and leakage rate testing.

The examination requirements of ASME XI, Subsection IWL, are essentially identical to those contained in Regulatory Guide 1.35, Rev. 3, and are more rigorous than those required by 10 CFR 50, Appendix J and Regulatory Guide 1.163. Previous visual examinations of containment concrete and post-

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

tensioning system surfaces have not revealed any indications of abnormal degradation of the containment. The five-year frequency for IWL examinations is adequate in lieu of the general visual examination frequency specified in Regulatory Guide 1.163 for containment concrete and post-tensioning system examinations.

The ability of the containment to perform its design function will not be impaired by the implementation of this amendment at Oconee Nuclear Station. Consequently, no safety margins will be impacted.

Duke has concluded based on this information there are no significant hazards considerations involved in this amendment request.

ATTACHMENT 5

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10CFR51.22 (b), an evaluation of the proposed amendments has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the No Significant Hazards Consideration Evaluation which is contained in Attachment 4.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   This amendment will not significantly change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

   This amendment will not significantly increase the individual or cumulative occupation radiation exposure.

In summary, this amendment request meets the criteria set forth in 10CFR51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.