50-269 Superseded Per Reissued ONS Select License Commitments manual Std 9/13/99 #9909160193

Oconee Nuclear Station
Selected Licensee Commitments
List of Effective Pages

Page	Revision Date
LOEP 1	6/30/99
LOEP 2	5/11/99
LOEP 3	4/12/99
LOEP 4	6/29/99
LOEP 5	6/24/99
LOEP 6	6/24/99
LOEP 7	6/16/99
LOEP 8	4/29/99
LOEP 9	6/30/99
LOEP 10	4/29/99
16.0-1	5/11/99
16.0-2	3/27/99
16.0-3	3/27/99
16.0-4	5/10/99
16.0-5	5/10/99
16.0-6	5/10/99
16.1-1	3/27/99
16.2-1	3/27/99
16.2-2	3/27/99
16.2-3	3/27/99
16.3-1	3/27/99
16.5.1-1	3/27/99
16.5.1-2	3/27/99
16.5.2-1	5/11/99
16.5.2-2	5/11/99
16.5.2-3	5/11/99
16.5.2-4	Delete 5/11/99
16.5.2-5	Delete 5/11/99
16.5.3-1	3/27/99
16.5.3-2	3/27/99
16.5.3-3	3/27/99
16.5.4-1	3/27/99
16.5.5-1	3/27/99
16.5.6-1	3/27/99
16.5.7-1	3/27/99
16.5.7-2	3/27/99

Oconee Nuclear Station
Selected Licensee Commitments
List of Effective Pages

Oconee Nuclear Station
Selected Licensee Commitments
List of Effective Pages

Page	Revision Date
16.6.4-3	3/27/99
16.6.4-4	3/27/99
16.6.4-5	3/27/99
16.6.4-6	3/27/99
16.6.5-1	3/27/99
16.6.6-1	3/27/99
16.6.7-1	3/27/99
16.6.8-1	3/27/99
16.6.9-1	3/27/99
16.6.9-2	3/27/99
16.6.10-1	3/27/99
16.6.10-2	3/27/99
16.6.10-3	3/27/99
16.6.12-1	3/27/99
16.6.12-2	3/27/99
16.6.12-3	3/27/99
16.6.12-4	3/27/99
16.6.12-5	3/27/99
16.6.12-6	3/27/99
16.6.12-7	3/27/99
16.7.1-1	3/27/99
16.7.1-2	3/27/99
16.7.2-1	3/27/99
16.7.2-2	3/27/99
16.7.2-3	3/27/99
16.7.3-1	3/27/99
16.7.3-2	3/27/99
16.7.4-1	3/27/99
16.7.5-1	3/27/99
16.7.5-2	3/27/99
16.7.5-3	3/27/99
16.7.5-4	3/27/99
16.7.6-1	3/27/99
16.7.7-1	3/27/99
16.7.7-2	3/27/99
16.7.8-1	3/27/99

Oconee Nuclear Station
Selected Licensee Commitments
List of Effective Pages

Page	Revision Date
16.8.6-3	3/27/99
16.8.7-1	3/27/99
16.8.8-1	3/27/99
16.9.1-1	6/24/99
16.9.1-2	3/27/99
16.9.1-3	3/27/99
16.9.1-4	3/27/99
16.9.1-5	3/27/99
16.9.2-1	3/27/99
16.9.2-2	3/27/99
16.9.2-3	3/27/99
16.9.2-4	3/27/99
16.9.3-1	3/27/99
16.9.3-2	3/27/99
16.9.4-1	3/27/99
16.9.4-2	3/27/99
16.9.4-3	3/27/99
16.9.4-4	3/27/99
16.9.4-5	3/27/99
16.9.5-1	4/29/99
16.9.5-2	4/29/99
16.9.5-3	4/29/99
16.9.5-4	4/29/99
16.9.6-1	3/27/99
16.9.6-2	3/27/99
16.9.6-3	3/27/99
16.9.6-4	3/27/99
16.9.6-5	3/27/99
16.9.6-6	3/27/99
16.9.6-7	3/27/99
16.9.6-8	3/27/99
16.9.6-9	3/27/99
16.9.7-1	6/24/99
16.9.7-2	6/24/99
16.9.7-3	6/24/99
16.9.7-4	6/24/99
16.9.7-5	Delete

Oconee Nuclear Station
Selected Licensee Commitments
List of Effective Pages

Page	Revision Date
16.9.7-6	Delete
16.9.7-7	Delete
16.9.7-8	Delete
16.9.7-9	Delete
16.9.8-1	6/24/99
16.9.8-2	Delete
16.9.8-3	Delete
16.9.8-4	Delete
16.9.8-5	Delete
16.9.8-6	Delete
16.9.8-7	Delete
16.9.8a-1	3/27/99
16.9.8a-2	3/27/99
16.9.8a-3	3/27/99
16.9.9-1	3/27/99
16.9.9-2	3/27/99
16.9.10-1	3/27/99
16.9.10-2	3/27/99
16.9.11-1	6/16/99
16.9.11-2	6/16/99
16.9.11-3	6/16/99
16.9.11-4	6/16/99
16.9.11-5	6/16/99
16.9.11-6	6/16/99
16.9.11-7	6/16/99
16.9.12-1	5/10/99
16.9.12-2	5/10/99
16.9.12-3	5/10/99
16.9.12-4	5/10/99
16.9.12-5	5/10/99
16.9.12-6	5/10/99
16.9.12-7	5/10/99
16.9.12-8	5/10/99
16.9.13-1	3/27/99
16.9.14-1	3/27/99
16.9.15-1	3/27/99
16.9.15-2	3/27/99

Page	Revision Date
16.9.15-3	3/27/99
16.9.16-1	3/27/99
16.9.16-2	3/27/99
16.9.16-3	3/27/99
16.9.17-1	3/27/99
16.9.17-2	3/27/99
16.9.18-1	3/27/99
16.9.18-2	3/27/99
16.9.18-3	3/27/99
16.9.18-4	3/27/99
16.9.18-5	3/27/99
16.9.18-6	3/27/99
16.9.18-7	3/27/99
16.9.18-8	3/27/99
16.10.1-1	3/27/99
16.10.1-2	3/27/99
16.10.1-3	3/27/99
16.10.2-1	3/27/99
16.10.3-1	3/27/99
16.10.3-2	3/27/99
16.10.4-1	3/27/99
16.10.5-1	3/27/99
16.10.6-1	3/27/99
16.10.7-1	4/29/99
16.10.7-2	4/29/99
16.10.7-3	4/29/99
16.10.7-4	4/29/99
16.10.7-5	4/29/99
16.10.7-6	4/29/99
16.10.7-7	4/29/99
16.10.7-8	4/29/99
16.10.7-9	4/29/99
16.11.1-1	3/27/99
16.11.1-2	3/27/99
16.11.1-3	3/27/99
16.11.1-4	3/27/99
16.11.1-5	3/27/99

Page	Revision Date
16.11.1-6	3/27/99
16.11.1-7	3/27/99
16.11.2-1	3/27/99
16.11.2-2	3/27/99
16.11.2-3	3/27/99
16.11.2-4	3/27/99
16.11.2-5	3/27/99
16.11.2-6	3/27/99
16.11.3-1	3/27/99
16.11.3-2	3/27/99
16.11.3-3	3/27/99
16.11.3-4	3/27/99
16.11.3-5	3/27/99
16.11.3-6	3/27/99
16.11.3-7	3/27/99
16.11.3-8	3/27/99
16.11.3-9	3/27/99
16.11.3-10	3/27/99
16.11.3-11	3/27/99
16.11.3-12	3/27/99
16.11.3-13	3/27/99
16.11.3-14	3/27/99
16.11.3-15	3/27/99
16.11.3-16	3/27/99
16.11.3-17	3/27/99
16.11.3-18	3/27/99
16.11.4-1	3/27/99
16.11.4-2	3/27/99
16.11.4-3	3/27/99
16.11.4-4	3/27/99
16.11.4-5	3/27/99
16.11.4-6	3/27/99
16.11.5-1	3/27/99
16.11.5-2	3/27/99
16.11.5-3	3/27/99
16.11.6-1	3/27/99

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

16.0 SELECTED LICENSEE COMMITMENTS

16.1 INTRODUCTION

This chapter provides a single location in the UFSAR where certain selected licensee commitments are presented. The content of this chapter is based on the results of application of a set of criteria to determine the content of technical specifications. For purposes of administrative ease, this chapter is maintained in a separate manual, The Oconee Nuclear Station Selected Licensee Commitments Manual. Those previous technical specification requirements which did not meet the criteria are relocated in this chapter.

The control of the Oconee Nuclear Station selected licensee commitment program and manual shall be in accordance with approved directive NSD 221 Facility Operating License and Technical Specifications Amendments/Selected Licensee Commitments/Technical Specifications Bases Changes. The manual is officially designated as Chapter 16 of the Oconee UFSAR. The original issue and subsequent revisions of the manual are approved by the station manager or his designee. Administrative requirements of the manual are the responsibility of the Site Regulatory Compliance Section.

Changes to these Selected Licensee Commitments may be made, pursuant to 10CFR50.59, only after the bases for the requirement have been clearly established and after a multidisciplinary review by Qualified Reviewers, including onsite Operations personnel.

Additional NRC commitments, as selected by the Station manager or designee may be located in this chapter. It is the intent of this chapter to provide information regarding systems that are a part of the licensing basis, as described in the UFSAR, but are not of such a level of importance that they need to be under the rigorous control provided by technical specifications.

This chapter includes Surveillance Requirements for certain systems, and remedial actions to be taken in the event the system is inoperable. A bases for the commitment is also provided. Reference is also provided to specific sections of the UFSAR where the information relative to the commitment is further described.

16.2 APPLICABILITY

This section provides the general requirements applicable to each of the Commitments within Section 16.0, Selected Licensee Commitments.

16.2.1 Commitments shall be met during the MODES or other specified conditions in the Applicability.

16.2.2 Upon discovery of a failure to meet a Commitment, the associated Actions shall be met. If the commitment is met or no longer required prior to expiration of the specified time intervals, completion of the Actions is not required.

16.2.3 When a Commitment is not met, except as provided in the associated Actions, the Station Manager and/or responsible Group Superintendent will determine any further actions.

Where corrective measures are completed that permit operation under the Actions, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Commitment. Exceptions to these requirements are stated in the individual Commitments.

16.2.4 Entry into a MODE other specified condition in the Applicability must be made with: (1) the full complement of required systems, equipment, or components OPERABLE, and (2) all other parameters as specified in the Commitment being met without regard for allowable deviations and out-of-service provisions contained in the Actions.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded without the approval of the Station Manager and/or the responsible Group Superintendent.

Exceptions to this provision have been provided for a limited number of Commitments when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the Actions of the appropriate Commitments.

When a commitment is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated Action to be entered permits continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time unless approved by the Station manager and/or responsible Group Superintendent. This commitment shall not prevent changes in MODE or other conditions specified in the Applicability that are required to comply with Actions. Exception to this commitment are stated in the individual Commitment.

16.2.5 Commitments, including the associated Actions, shall apply to each individually unless otherwise indicated as follows:

Whenever the Commitment refers to systems or components which are shared by units, the Actions will apply to all affected units simultaneously.

Whenever the Commitment applies to only one unit, this will be identified in the commitment section of the Commitment; and

Whenever certain portions of a Commitment contain operating parameters, setpoints, etc. which are different for each unit, this will be identified in parentheses or footnotes.

16.2.6 Surveillance Requirements shall be performed during the MODE or other specified condition in the Applicability for individual Commitments unless otherwise stated in an individual Surveillance Requirement or reference.

Failure to meet a Surveillance Requirement, whether such failure is experienced during the performance of the Surveillance Requirement or between performances of the Surveillance Requirement, shall be failure to meet the commitment. Failure to perform a Surveillance Requirement within the Specified Frequency shall be failure to meet the Commitment except as provided in Commitment 16.2.8. Surveillance Requirements do not have to be performed on inoperable equipment or variables outside specified limits.

Surveillance Requirements are necessary to ensure the Commitments are met and will be performed during the MODE or other specified condition in the Applicability. Provisions for additional Surveillance Requirements to be performed without regard to the applicable operational condition or MODE or other specified conditions in the Applicability are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Conditions need only be performed when the Special Condition is being utilized as an exception to an individual Commitment.

16.2.7 Each Surveillance Requirement shall be performed on its specified frequency with a maximum allowable extension not to exceed 50% of the test frequency, unless specified differently in the individual Commitment.

Allowable tolerances are provided for performing Surveillance Requirements beyond those specified in the nominal testing frequency. The tolerance is necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a testing frequency does not negate this allowable tolerance value, and permits the performance of more frequent surveillance activities. For frequencies specified as "once" the above extension does not apply to the first performance.

16.2.8 Exceptions to these requirements are stated in the individual Commitments or may be approved by the Station Manager and/or responsible Group Superintendent.

16.2.9 Surveillance Requirements associated with a Commitment shall be performed within the specified time interval prior to entry into a MODE or other specified condition in the Applicability. The intent of this provision is to ensure that Surveillance Requirement have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Commitment unless otherwise approved by the Station manager and/or the responsible Group Superintendent.

Entry into a MODE or other specified condition in the Applicability shall not be made unless the Surveillance Requirement associated with the Commitment have been performed within the specified frequency or as approved by the Station Manager and/or responsible Group Superintendent. These provisions shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with Actions.

16.2.10 Surveillance Requirement shall apply to each unit individually unless otherwise indicated or whenever certain portions of a specification contain testing parameters different for each unit, which will he identified in notes.

16.3 DEFINITIONS

The definitions in the Oconee Technical Specifications apply to defined terms used herein. The following additional defined terms are applicable throughout this Selected Licensee Commitment document:

16.3.1 DELETED

16.3.2 SOLIDIFICATION - SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrations as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the Process Control Program (PCP).

16.3.3 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

16.3.4 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment System components.

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.1 Reactor Coolant System Vents

COMMITMENT

a. The following reactor coolant system vent paths shall be OPERABLE:

1) Reactor Vessel Head Vent
2) Pressurizer Steam Space Vent (through PORV)
3) RCS Loop A High Point Vent
4) RCS Loop B High Point Vent

b. For each vent path, two electrically-operated valves shall be capable of being opened, and all manual valves shall be open.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS vent path inoperable.	A.1 Restore to OPERABLE status.	30 days
B. Two or more RCS vent paths inoperable.	B.1 Restore to OPERABLE status.	72 hours
C. The Required Actions and associated Completion Times of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.5.1.1	Verify an open flow path for each RCS vent path by testing the head vents and loop high point vents.	18 months
SR 16.5.1.2	Perform high point vent valve testing.	In accordance with ASME Section XI

BASES

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The RCS vents have two valves in series which are capable of being powered from emergency buses. The valves are normally closed with power removed to prevent inadvertent opening of the valves. In order for a vent path to perform its intended safety function of venting, the two electrically-operated valves in the flow path must be capable of being opened, and all manual valves must be open.

REFERENCES

1. NUREG 0737, Item II.B.1

2. Generic Letter 83-37

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.2 Low Temperature Overpressure Protection System

COMMITMENT Perform SURVEILLANCE REQUIREMENTS.

APPLICABILITY: MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODES 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Not Applicable	A. Not Applicable	Not Applicable

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.5.2.1	Verify travel stops limit flow through HP-120 to the specified limit.	18 months
SR 16.5.2.2	Perform Channel Calibration on pressurizer level and RCS pressure alarms.	18 months
SR 16.5.2.3	Perform an inspection of the PORV.	every 2 refueling cycles

SURVEILLANCE		FREQUENCY
SR 16.5.2.4	-------------------NOTE-------------------- Only required to be met when vent(s) are being used for overpressure protection. --- Verify valves in the flowpath for the RCS vent(s) are open.	12 hours for valves not locked, sealed, or otherwise secured open <u>AND</u> 31 days for valves locked, sealed, or otherwise secured open

BASES

BACKGROUND

This SLC is provided to establish the SURVEILLANCE REQUIREMENTS for operability of the Administrative Controls (second train) of the LTOP system in accordance with ITS 3.4.12.

The COMMITMENTS for the Administrative Controls are located in the LCO Section of ITS Bases 3.4.12. The associated ACTION requirements are contained in ITS 3.4.12. The BASES for the LTOP requirements and the associated Administrative Controls are described in ITS Bases 3.4.12.

APPLICABILITY

The SLC is applicable when the provisions of ITS 3.4.12 are applicable. This SLC is not applicable for operating conditions above 325°F since the possibility of non-ductile failure is significantly diminished. Vent paths capable of mitigating the most limiting LTOP event are specified in Operations procedures. If an LTOP event were to occur, violation of this SLC could result in exceeding the brittle fracture pressure limits, overstressing the reactor vessel and closure head, or require reanalysis to demonstrate the resulting stresses would not impair further operation.

SURVEILLANCE REQUIREMENTS

The identified surveillance requirements are provided to assure that the second train of LTOP is functioning properly and gives the operator 10 minutes to mitigate an LTOP event.

REFERENCES:

1. ITS 3.4.12
2. 10 CFR 50 Appendix G "Fracture Toughness Requirements."
3. Calc. File OSC-4445, Revision 8, "Low Temperature Overpressure Protection Evaluation," dated 4/30/99.

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.3 Loss of Decay Heat Removal

COMMITMENT The following conditions shall be met:

a. Conduct a containment closure survey to identify containment penetrations that would need to be closed in the event of a loss of decay heat removal capability and to ensure that containment closure can be achieved within 2 1/2 hours,

b. Two operable core exit thermocouple indications and alarm shall be available. The core exit temperature shall be monitored and recorded at least once every two hours,

c. The LT-5 Reactor vessel level indication system shall be available and operable.

d. An ultrasonic Reactor vessel level detection system, or other backup level indicating system, shall be available and operable in addition to LT-5.

e. Both Main Feeder Buses (MFB) shall be energized.

f. Two sources of power shall be available to supply the Main Feeder buses.

g. Two of the following means of adding inventory to the RCS are available and operable:

1. A gravity flow path from the BWST

2. One Bleed Transfer Pump (BTP) and connecting piping

3. A High Pressure Injection (HPI) pump

h. Both steam generators upper primary side handhole covers, or equivalent RCS vent path, shall be removed.

i. Testing and maintenance activities shall be reviewed to ensure no adverse effects on systems and components required for decay heat removal. Those activities which pose a substantial threat to decay heat removal capability will be prohibited.

---------------------------NOTES---------------------------

1. Commitment b is required to be met only when the reactor vessel head is in place.
2. Commitment d is a Duke Power internal commitment, not a NRC commitment.

APPLICABILITY: MODE 5 and 6 with RCS level < 50 inches above centerline of the reactor vessel hot leg and with irradiated fuel in the reactor vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more commitment not met.	A.1 As determined by Station Manager or Responsible Group Superintendent.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.5.3.1 NA	NA

BASES

Generic Letter 88-17, Loss of Decay Heat Removal, was issued October 17, 1988 by the Nuclear Regulatory Commission (NRC) due to numerous events in the industry involving a loss of decay heat removal capability while the reactor vessel was in a drained down condition. The major concern is that substantial

core decay heat may pose a significant likelihood of a release due to a severe core damage accident.

Babcock and Wilcox (B&W) designed plants are not as sensitive to the loss of decay heat removal problems, but the additional actions of this commitment provide added assurance of avoiding possible problems while the reactor vessel is in a reduced inventory condition.

<u>REFERENCES</u>:

1. Generic Letter 88-17

2. ONS responses to Generic Letter 88-17, dated January 3,1989 and February 2, 1989.

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.4 ----DELETED-----

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.5 -----DELETED-----

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.6 -----DELETED-----

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.7 Chemistry Requirements

COMMITMENT

The concentration of oxygen, chloride and fluoride in the RCS and pressurizer shall be maintained within limits.

-------------------------NOTE-------------------------
Prior to exceeding 525°F, the limits of SR 16.5.7.1 shall be met.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more chemistry limits not met in MODE 3.	A.1	Initiate action to restore to within limits.	8 hours
	<u>AND</u>		
	A.2	Restore to within limits.	24 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 5.	24 hours
C. One or more chemistry limits not met in MODE 1 or MODE 2.	C.1	Initiate action to restore to within limits.	8 hours
	<u>AND</u>		
	C.2	Restore to within limits.	24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Actions and associated Completion Times for Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	24 hours 72 hours
E. RCS oxygen level exceeds 1.00 ppm. <u>OR</u> RCS chloride or fluoride level exceeds 1.50 ppm.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.	8 hours 24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 16.5.7.1	-------------------NOTES------------------ 1. Not applicable to oxygen concentration in the pressurizer in MODE 3 with RCS temperature ≤ 525°F. 2. Not required to be performed for the pressurizer. -- Verify fluoride, chloride and oxygen concentrations in the RCS and the pressurizer are ≤ 0.15 ppm, ≤ 0.15 ppm, and ≤ 0.10 ppm, respectively.	three times per week
SR 16.5.7.2	-------------------NOTE------------------- Not required to be performed for the pressurizer. -- Verify oxygen concentration in the pressurizer is ≤ 0.20 ppm.	N/A
SR 16.5.7.3	Analyze the RCS tritium level.	92 days.

BASES

BACKGROUND

This commitment incorporates the changes to the Technical Specifications that ONS proposed in a letter dated February 20, 1996. The Technical Specifications revision, approved on February 19, 1997, removed several chemistry requirements from the Oconee Technical Specifications. The limits and surveillance for the oxygen, chloride, and fluoride concentration levels were relocated to the SLC as stated in the proposed amendment. In addition, the relocation of the tritium analysis from the Technical Specifications to the SLCs was covered in the proposed amendment.

APPLICABLE SAFETY ANALYSIS

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within limits, the integrity of the reactor coolant system is protected against potential stress corrosion attack.

Oxygen is normally expected to be below detectable limits. Oxygen is normally below limits since dissolved hydrogen is used when the reactor is critical and a residual of hydrazine is used in the pressurizer when the reactor is subcritical to control the oxygen. The requirement that the oxygen concentration not exceed 0.1 ppm in the RCS when in MODES 1 and 2 and when in MODE 3 when exceeding 525°F provides added assurance that stress corrosion cracks will not occur.

During the startup of an Oconee unit the Oconee unit will be held at MODE 4 for initial cleanup. The control of the chloride and fluoride is accomplished through the use of purification demineralizers and it is essential that this control be established during startup since it cannot be reasonably expected that normal power operation will change their performance in any way. This approach, which is recommended by the EPRI PWR Primary Water Chemistry Guidelines, during startup reduces the possible negative effects on material integrity at high temperatures by establishing these specifications at reasonably achievable low levels before entering MODE 3.

APPLICABILITY

SLC 16.5.7 is applicable in MODES 1, 2, and 3.

During heatup from MODE 5, oxygen control is accomplished through the addition of hydrazine to the pressuirizer or hydrogen to the RCS. The reduction of pressurizer oxygen during heatup is influenced by the following considerations.

a. Venting of the pressurizer during startups below 600 psig is not permitted due to the low range pressure transmitter for the RCS being on the same line.

b. Pressurizer volume is small compared to the total RCS volume.

c. Influx of water with oxygen scavenging hydrazine is limited by the operating constraints placed on the pressurizer to control system pressure.

Significantly higher temperatures exist in the pressurizer than in the reactor coolant system in the preliminary heatup. Because of the problems associated with oxygen scavenging in the pressurizer, a maximum of 0.20 ppm oxygen (twice the specified normal power operation RCS concentration) is allowed prior to entering MODE 3 and time constraints are placed on operating at this concentration in MODES 1, 2, and 3 to ensure all possible effort is made to bring the value below 0.10 ppm as rapidly as reasonably achievable.

ACTIONS

A.1 and A.2

If one or more chemistry limits are not met in MODE 3, actions will be initiated within 8 hours to restore the chemistry levels to within the commitment requirements. The RCS chemistry levels shall be restored to within the commitment requirements within 24 hours.

B.1 and B.2

If the Required Action and associated Completion Time for Condition A are not met, the unit shall be placed in MODE 5 within 24 hours.

C.1

If the oxygen, chloride, or fluoride limits are exceeded in MODES 1 and 2, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, increase the hydrogen concentration in the makeup tank, etc.). The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time. Thus, the period of eight hours to initiate corrective action and the period of 24 hours to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analyses are correct and to locate the source of contamination.

D.1 and D.2

If the Required Action and associated Completion Time for Condition C are not met, the unit shall be placed in MODE 3 within 24 hours and MODE 5 within 72 hours.

E.1 and E.2

If the RCS chemistry levels exceed 1.00 ppm for oxygen and 1.50 ppm for chloride and fluoride, the unit shall be placed in MODE 3 within 8 hours and

MODE 5 within 24 hours. This requirement is consistent with the action level 3 requirements in the EPRI PWR Primary Water Chemistry Guidelines.

SURVEILLANCE REQUIREMENTS

SR 16.5.7.1

This SR ensures that the RCS chemistry levels are within the SLC requirements. The 3 time per week Frequency is considered adequate to provide such assurance.

SR 16.5.7.2

This SR establishes the limit for oxygen concentration in the pressurizer. No specific frequency is required and the Note states the SR is not required to be performed. Chemistry procedures provide appropriate controls for sampling the pressurizer. This presentation establishes the limit for this parameter without requiring actual performance of the SR.

SR 16.5.7.3

This SR analyzes the tritium level every 92 days.

REFERENCES

1. UFSAR, Section 5.2.1.7
2. UFSAR, Section 9.3.1-2
3. Stress Corrosion of Metals, Logan
4. Corrosion and Wear Handbook, O. J. DePaul, Editor
5. EPRI PWR Primary Water Chemistry Guidelines
6. Duke letters to the NRC dated February 20, 1996, and October 16. 1996.
7. NRC Safety Evaluation dated February 19, 1997.

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.8 Pressurizer

COMMITMENT

a. The pressurizer shall have a steam bubble and a water level > 80 inches in MODES 1 and 2.

b. The pressurizer heatup and cooldown rates shall be ≤ 100°F/hr.

c. The pressurizer spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 410°F.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.5.8.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.1.2.6 and 3.1.3.4 during the conversion to ITS.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a startup accident.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1699 and BAW-1697.

The spray temperature difference is imposed to maintain the thermal stresses at the pressurized spray line nozzle below the design limit.

REFERENCES

UFSAR Section 15.3

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.9 Testing Following Opening of System (Core Barrel Bolt Inspections)

COMMITMENT: Two sets of main internal bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
16.5.9.1	Visually inspect the core barrel to lower grid cylinder welded bolt locking caps.	Whenever the internals are removed from the vessel
16.5.9.2	Visually inspect the core barrel to core support shield welded bolt locking caps.	------NOTE----- The provisions of SLC 16.2.7 do not apply. -------------- 18 months +25% AND Whenever the internals are removed from the vessel

BASES

The requirement(s) of this SLC section were relocated from CTS 4.2.2 during the conversion to ITS.

To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) must remain in place and under tension. This is verified by visual inspection to determine that the welded bolt locking caps remain in place.

REFERENCES

N/A

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.10 Loss of Reactor Coolant

COMMITMENT Loss of reactor coolant through reactor coolant pump (RCP) seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system (RCS), shall be ≤ 30 gpm when added to RCS LEAKAGE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS Leakage evaluated as unsafe. OR Loss of reactor coolant through RCP seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the RCS is > limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.5.10.1 Initiate evaluation of safety implications of RCS leakage.	Once within 4 hours of detection

BASES

The requirement(s) of this SLC section were relocated from the CTS.1.6.2, 3.1.6.6, and 3.1.6.9 during the conversion to ITS

Water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks (Refs. 1, 2, and 4). Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small breaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

The upper limit of 30 gpm, which includes RCS LEAKAGE (Ref. 3), is based on the contingency of a complete loss of station power. A 30 gpm loss of water in conjunction with a complete loss of station power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore electrical power to the station and makeup flow to the reactor coolant system (Ref. 2).

REFERENCES

1. UFSAR, Section 3.1.16 and 3.1.17.

2. UFSAR, Section 5.2.3.10.3 and 5.2.3.10.5.

3. ITS 3.4.13, "RCS Operational LEAKAGE."

4. Generic Letter 88-05, "BORIC ACID CORROSION OF CARBON STEEL REACTOR PRESSURE BOUNDARY COMPONENTS IN PWR PLANTS," dated March 17, 1988.

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.11 Subcriticallity

COMMITMENT The reactor shall be maintained subcritical by an amount greater than or equal to the calculated reactivity insertion due to depressurization.

APPLICABILITY: MODE 2 and 3 with Tavg < 525°F
MODE 4, 5, and 6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.5.11.1 N/A	N/A

<u>BASES</u>

The requirement(s) of this SLC section were relocated from CTS 3.1.3.3 during the conversion to ITS.

The potential reactivity insertion due to the moderator pressure coefficient [2] that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1% $\Delta k/k$.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

<u>REFERENCES</u>

N/A

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.12 RCS LEAK TESTING FOLLOWING OPENING OF SYSTEM

COMMITMENT Perform specified SR.

APPLICABILITY: MODE 1,
MODE 2 with Keff $\geq$ 1.0

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.5.12.1	Perform RCS leakage test at not less than 2200 psig.	Once following any opening of RCS

BASES

The requirement(s) of this SLC section were relocated from CTS 4.3.2 during the conversion to ITS.

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes. The specific code and edition thereof shall be consistent with 10 CFR.55a.

REFERENCES

N/A

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.13 High Pressure Injection (HPI) and the Chemical Addition Systems

COMMITMENT

a. Two high pressure injection pumps per unit shall be OPERABLE except as specified in ITS 3.5.2.

b. The concentrated boric acid storage tank (CBAST) shall be OPERABLE.

APPLICABILITY: MODE 1,
MODE 2 with Keff $\geq$ 1.0

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CBAST inoperable. AND BWST OPERABLE.	A.1 Restore CBAST to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with Keff < 1.0.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.5.13.1	Verify boron concentration in CBAST is within the limit specified in the COLR.	7 days

BASES

The requirement(s) of this SLC section were relocated from CTS 3.2 and Table 4.1-3, Item 6 during the conversion to ITS.

One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and OPERABLE.

This source shall be the concentrated boric acid storage tank (CBAST). The CBAST is OPERABLE when volume and boron concentration are within the limits of the Core Operating Limits Report (COLR) with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be OPERABLE and shall have the same temperature requirement as the CBAST. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be OPERABLE.

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration. This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the CBAST or a bleed transfer pump aligned to take suction from the CBAST. The boric acid pump associated with the CBAST is normally used for small additions during operation and the bleed transfer pumps are utilized when larger volumes are to be added. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank (BWST). Part a to the commitment is relocated from CTS 3.2.2 for completeness (i.e., verbatim relocation). The requirement for two HPI pumps except as specified in ITS 3.5.2 establishes no additional requirements other than those specified in ITS 3.5.2. Compliance with ITS 3.5.2 (LCO and ACTIONS) for HPI pumps establishes compliance with this SLC for commitment part a.

The quantity of boric acid in storage in the CBAST or the BWST is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at 60°F with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit are analyzed with the limits presented in the COLR. The cycle specific analyses determine the volume and boron concentration requirements for the BWST and CBAST necessary to borate to a cold shutdown condition (MODE 5). The volume requirements include a 10%

BASES (continued)

margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to MODE 5. One of the supplies requires the operability of the CBAST with an associated pump and flow path to ensure the capability to borate the RCS to MODE 5. This requirement is not one which must be immediately available since the shortest required timeframe to reach MODE 5 is 36 hours. Thus the required boric acid from the CBAST can be added by either of the bleed transfer pumps manually aligned to take suction from the CBAST and discharging to the inlet of the makeup filters at nominal flow rates of 100 gpm. Since there is sufficient time to make the alignment, manual alignment of the bleed transfer pumps is acceptable. This flow path and the associated pumps are equivalent from safety-related and seismic criteria to that of the CBAST pump and are capable of adding the required volume from the CBAST well within the minimum 36 hours. Equivalent volumes with lower concentrations will take longer than those volumes with higher concentrations, however, both can be added within the required timeframe.

During operation, the CBAST pump is normally aligned to the CBAST and discharges to the inlet of the makeup filters. Each CBAST pump is capable of delivering the required boric acid to the RCS within the required timeframe at a minimum flow of 7 gpm. Small volume additions from the CBAST will normally be added with the CBAST pump with the bleed transfer pumps being utilized for larger volume additions. An alternate method of addition is to inject boric acid from the BWST using the high pressure injection pumps.

The concentration of boron in the CBAST may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and the associated piping for the flowpaths will be kept at least 10°F above the crystallization temperature for the concentration present. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

1. UFSAR, Sections 9.3.1, and 9.3.2.
2. UFSAR, Figure 6-1.

16.6 ENGINEERED SAFETY FEATURES

16.6.1 Containment Leakage Tests

COMMITMENT — The local leak rate shall be measured for the containment penetrations listed in Table 16.6-1 in accordance with ITS SR 3.6.1.2.

APPLICABILITY — MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.6.1.1 NA	NA

BASES

This commitment establishes the list of penetrations that require local leak rate testing in accordance with ITS SR 3.6.1.2. This list was removed from the Technical Specifications in accordance with the guidance in NRC Generic Letter 91-08.

The requirement to leak test the blind isolation flanges on the containment Hydrogen Recombiner System permanent piping after each installation was relocated from CTS 4.4.3.1.b during conversion to the ITS.

REFERENCES

1. 10 CFR 50, Appendix J.
2. NRC Generic Letter 91-08.
3. UFSAR section 3.8.1.7.4, 6.2.3,and 6.2.4.

Table 16.6-1
List of Penetrations With 10 CFR 50 Appendix J Requirements

Penetration Number	System	Type A Test System Condition	Local Leak Test	Remarks
1	Pressurizer liquid sample line (Unit 1 only)	Note 1	Type C	Notes 2, 7b
2	OTSG A Sample line	Note 1	Type C	Note 7b
3	Component Cooling inlet line	Note 1	Type C	Note 3, 7d
4	OTSG B drain line	Note 1	None required	Note 7b
5a	RB normal sump drain line portion	Note 10	Type C	Note 7a, 7b, 9
5a	Hydrogen Recombiner drains portion	Note 10	Type C	Note 7a, 7d, 7e
5b	Post Accident Liquid Sample Line	Note 1	None Required	Note 2, 7c
6	Letdown line	Note 1	Type C	Note 2, 7b
7	RC Pump seal return line	Note 1	Type C	(Units 2 & 3) Note 7b, 9 (Unit 1) Note 3, 7b, 9
8a	Pressurizer Aux. Spray Line	Not Vented	None Required	Note 5, 7d
8b	Loop A nozzle warming line	Not Vented	None Required	Note 5, 7d
9	RCS normal makeup line and HP injection "A" loop	Not Vented	None Required	Note 5
10a	RC Pump B1 seal injection	Not Vented	Type C	Note 5, 7d, 9
10b	RC Pump B2 seal injection	Not Vented	Type C	Note 5, 7d, 9
11a	Fuel transfer tube cover	Not Vented	Type B	Note 6a, 11
11b	RC Makeup Pump suction	Note 1	Type C	Note 2
11c	Fuel transfer tube drain	Not Vented	Type C	Note 5
12a	Fuel transfer tube cover	Not Vented	Type B	Note 6a, 11
12b	RC Makeup Pump discharge	Note 1	Type C	Note 2
13	RB Spray inlet line	Not Vented	None Required	Note 5, 7d
14	RB Spray inlet line	Not Vented	None Required	Note 5, 7d
15	LPI and DHR inlet line	Not Vented	None Required	Note 4, 5
16	LPI and DHR inlet line	Not Vented	None Required	Note 4, 5
17	OTSG B Emergency FDW line	Not Vented	None Required	Note 5, 7d
18	Quench tank vent line	Note 1	Type C	Note 3, 7b, 9
19	RB purge inlet line	Vented	Type C	Note 7a, 7b, 9
20	RB purge outlet line	Vented	Type C	Note 7a, 7b, 9
21	LPSW to RC Pump motors and lube oil coolers inlet	Not Vented	None Required	Note 7b, 9

Table 16.6-1
List of Penetrations With 10 CFR 50 Appendix J Requirements

Penetration Number	System	Type A Test System Condition	Local Leak Test	Remarks
22	LPSW from RC Pump motors and lube oil coolers outlet	Not Vented	Type C	Note 7b
23a	RC Pump A1 seal injection	Not Vented	Type C	Note 5, 7d, 9
23b	RC Pump A2 seal injection	Not Vented	Type C	Note 5, 7d, 9
24a	RB H_2 Analyzer Train A	Vented	Type C	Note 7c
24b	RB H_2 Analyzer Train A	Vented	Type C	Note 7c
25	OTSG B Feedwater line	Not Vented	None Required	Note 7d, 14
26	OTSG A Main steam line	Not Vented	None Required	Note 5
27	OTSG A Feedwater line	Not Vented	None required	Note 7d, 14
28	OTSG B Main steam line	Not Vented	None required	Note 5
29	Quench tank drain line	Note 1	Type C	Note 3, 7b, 9
30, 31, 32	LPSW for RB Cooling units inlet line	Not Vented	None required	Note 5
33, 34, 35	LPSW for RB cooling units outlet line	Not Vented	None required	Note 5
36, 37	RB emergency sump recirculation line	Not Vented	None required	Note 5
38	Quench tank cooler inlet line	Note 1	Type C	Note 2, 7d
39a (Unit 2, 3 only)	CFT Vent Line	Note 1	None required	Note 3
39b	HP Nitrogen supply	Note 1	Type C	Note 2, 3
40	RB emergency sump drain line	Note 1	None required	
41	Instrument air supply & ILRT verification line	Vented	None required	Note 3
42a	RB H_2 Analyzer Train B	Vented	Type C	Note 7c
42b	RB H_2 Analyzer Train B	Vented	Type C	Note 7c
43	OTSG A drain line	Note 1	None required	Note 7b
44	Component cooling to control rod drive inlet line	Note 1	Type C	Note 3, 7d
45a	ILRT instrument line	Vented	Type C	Note 3, 7a
45b	ILRT instrument line	Vented	Type C	Note 3, 7a
45c (Units 2 & 3)	ILRT instrument line	Vented	Type C	Note 3, 7a
46	Reactor head-wash filtered water inlet	Note 1	Type C	Note 3, 9
47 (Unit 1 only)	Demineralized water supply to RC pump seal vents	Note 1	Type C	Note 3, 7d
48	Breathing air inlet	Vented	None required	Note 3

Table 16.6-1
List of Penetrations With 10 CFR 50 Appendix J Requirements

Penetration Number	System	Type A Test System Condition	Local Leak Test	Remarks
49 (Unit 1 only)	LP Nitrogen supply	Vented	None required	Note 3
50	OTSG A Emergency FDW line	Not Vented	None required	Note 5
51	ILRT Pressurization line	Vented	None required	Note 6a, 7a
52	HP injection to 'B' loop	Not Vented	None required	Note 5
53a (All)	HP Nitrogen supply to 'A' core flood tank	Note 1	Type C	Note 2, 3, 7d
53b (Units 2,3)	LP Nitrogen supply	Vented	None required	Note 2, 3, 7d
54	Component cooling outlet line	Note 1	Type C	Note 3, 7b, 9(8)
55	Demineralized water supply	Note 1	Type C	(Unit 1) Note 3 (Unit 2, 3) Note 3, 9
56	Spent fuel canal fill and drain	Note 1	None required	Note 3
57 (Unit 1 only)	DHR return line	Not Vented	None required	Note 4
58a (Unit 2, 3)	Pressurizer sample line	Note 1	Type C	Note 2, 7b
58b (All)	OTSG B sample line	Note 1	Type C	Note 7b
59	CF tank sample line	Note 1	None required	Note 2
60	RB sample line (outlet)	Note 1	Type C	Note 2, 7b, 9, 15
61	RB sample line (inlet)	Note 1	Type C	Note 2, 7b, 9. 15
62 (Units 2,3, Only)	DHR return line	Not Vented	None required	Note 4
90	Personnel hatch	Vented	Type B	Note 6b
91	Equipment hatch	Vented	Type B	Note 6c
92	Emergency hatch	Vented	Type B	Note 6b
101 through 105	Electrical Penetrations	Vented	Type B	Note 6a

NOTE 1 All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment atmosphere and to assure they will be subjected to the test differential pressure.

NOTE 2 Fluid system that is part of the reactor coolant pressure boundary or open directly to the containment atmosphere under post-accident conditions (vented to containment atmosphere during Type A test).

NOTE 3 Closed system inside containment that penetrates containment and postulated to rupture as a result of a loss of coolant accident (vented to containment atmosphere during Type A test).

NOTE 4 System required to maintain the plant in a safe condition during the test (need not be vented).

NOTE 5 System normally filled with water or under pressure and operating under post-accident condition (need not be vented).

Table 16.6-1
List of Penetrations With 10 CFR 50 Appendix J Requirements

NOTE 6

a. Containment penetration whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetration filled with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

b. Air lock door seals including door opening mechanisms which are part of the containment pressure boundary.

c. Doors with resilient seals or gaskets except for seal welded doors.

d. Components other than those above which must meet the acceptance criteria of Type B tests.

NOTE 7

a. Isolation valves provide a direct connection between the inside and outside atmosphere of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves.

b. Isolation valves are required to close automatically upon receipt of a containment isolation signal in response to controls intended to affect containment isolation.

c. Isolation valves are required to operate intermittently under post accident conditions.

d. Check valve(s) used for containment isolation.

e. Valves are normally closed but must be opened for hydrogen control.

NOTE 8 DELETED.

NOTE 9 Reverse direction test of inside containment isolation valve authorized. Leakage results are conservative.

NOTE 10 System is submerged during post-accident conditions and performance of Type A test. System will be drained to the extent possible.

NOTE 11 Type B test performed on the blind flanges inside the Reactor Building. Valves outside the containment are not tested.

NOTE 12 DELETED

NOTE 13 DELETED

NOTE 14 Closed system inside containment separated from the Reactor Coolant System and not postulated to rupture as a result of a loss of coolant accident.

NOTE 15 The blind isolation flanges on the Containment Hydrogen Recombiner System permanent piping shall be leak tested after each installation to ensure adequate isolation.

16.6 ENGINEERED SAFETY FEATURES

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
	OR	
	A.2.1 Verify that containment structural integrity is maintained, by performing an engineering evaluation of the containment structural integrity.	72 hours
	AND	
	A.2.2 Submit Tendon Surveillance Report in accordance with ITS 5.6.7.	30 days

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. -- 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.	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.

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

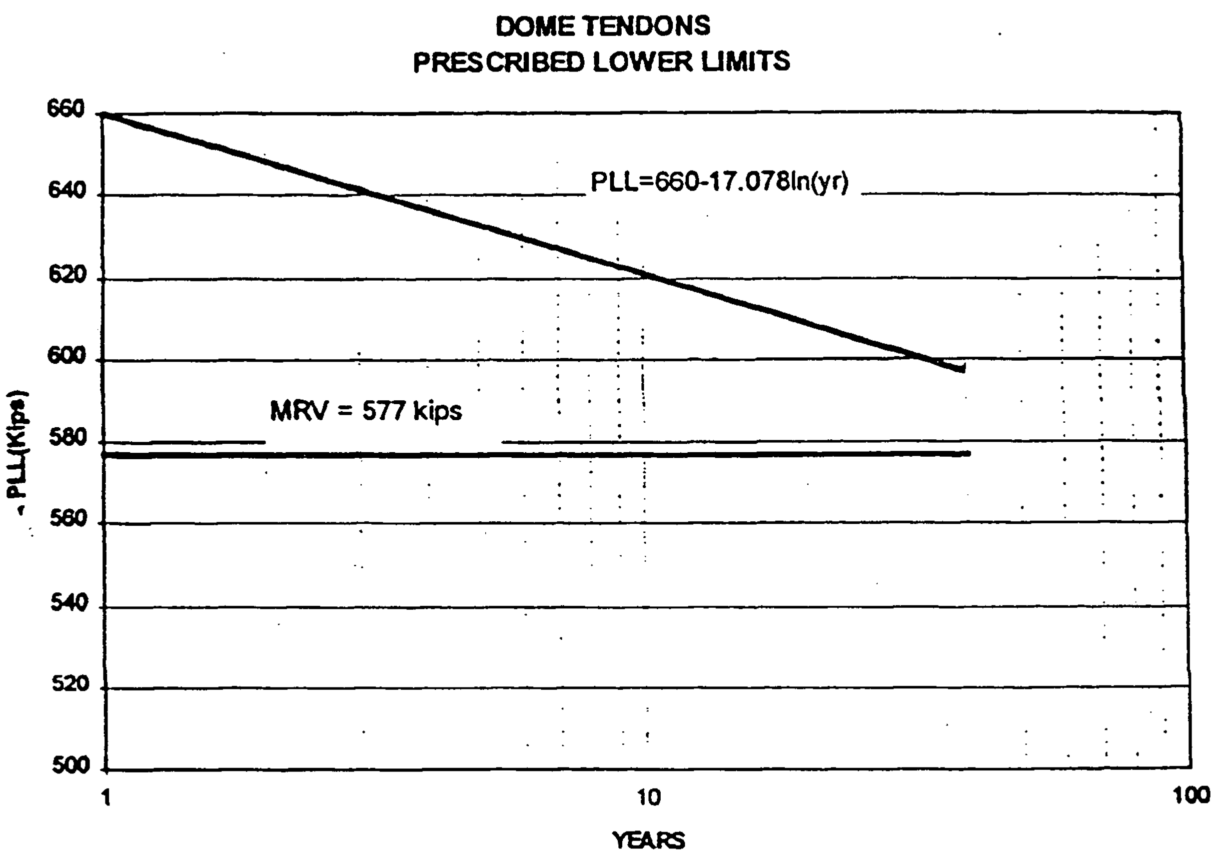


Figure 16.6.2-2
Hoop Tendon PLLs and MRVs

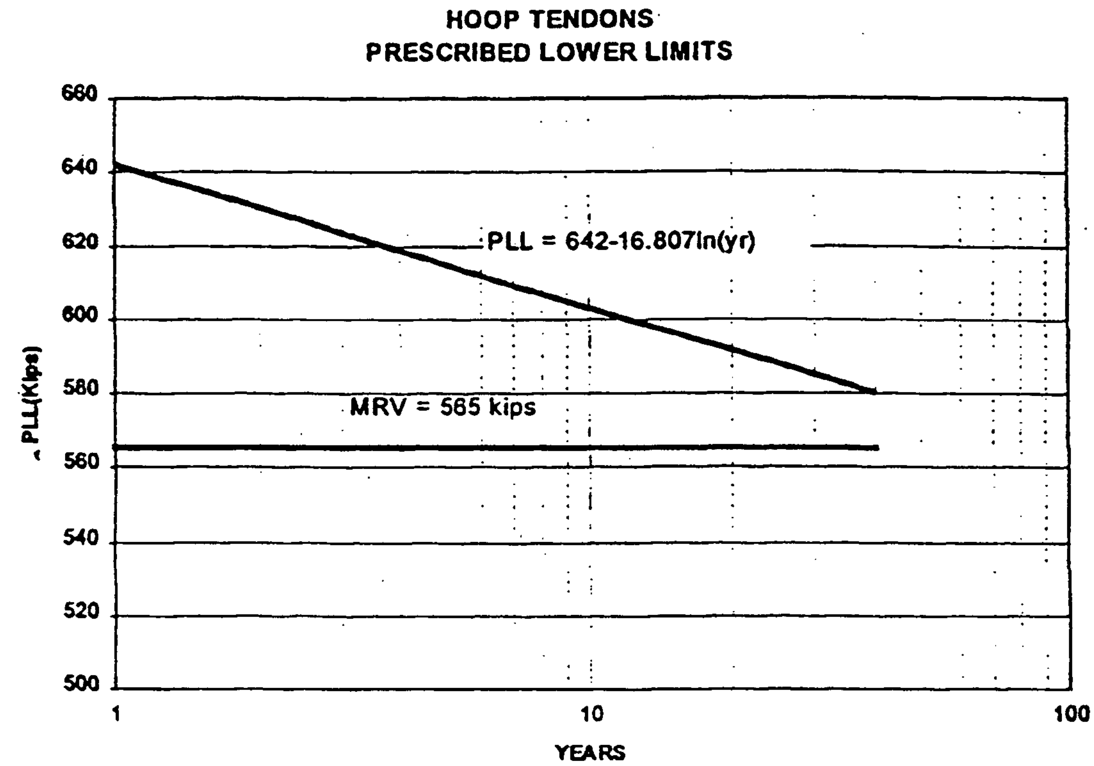
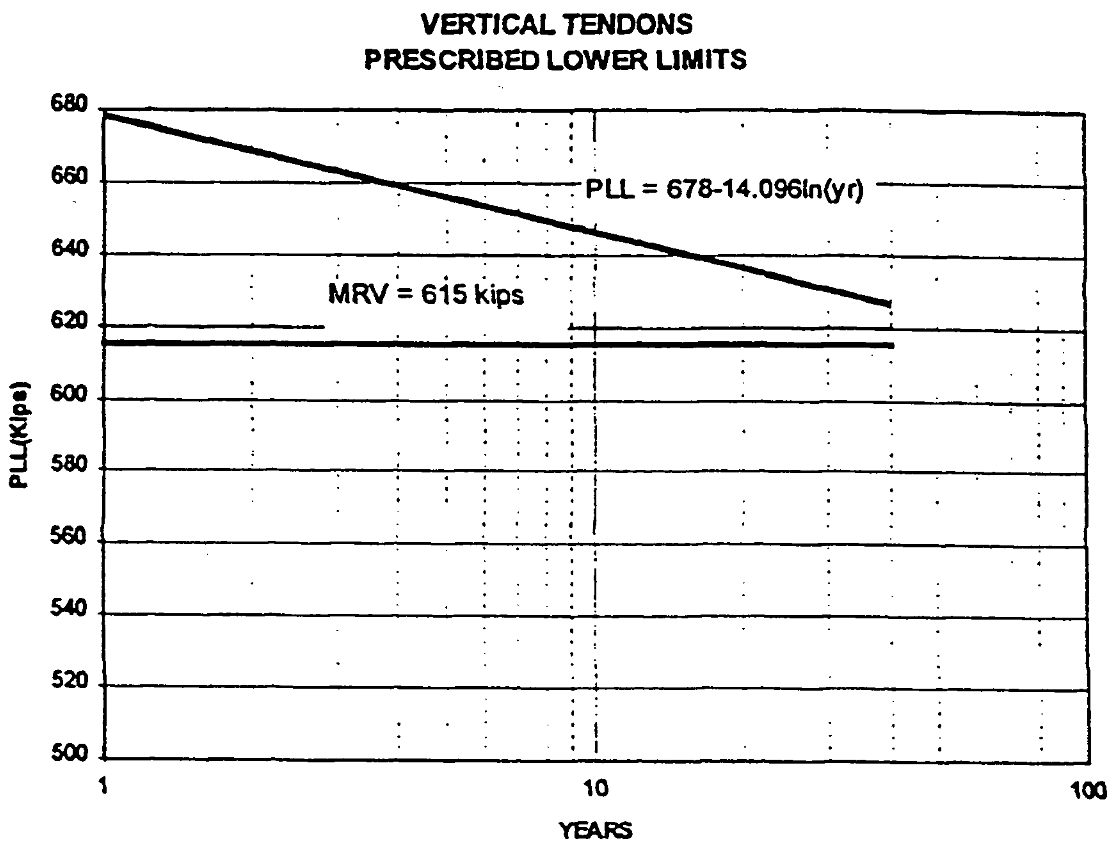


Figure 16.6.2-3
Vertical Tendon PLLs and MRVs



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.

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

16.6 ENGINEERED SAFETY FEATURES

16.6.3 Containment Heat Removal Verification Frequency

COMMITMENT Performed required SRs.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.3.1	Verify containment heat removal capability is sufficient to maintain post accident conditions within design limits.	As determined by LPI and RBCU fouling rate

BASES

The requirement(s) of this SLC section were relocated from CTS 4.5.3.1.b during the conversion to ITS.

The safety functions of the LPI system, RB Spray system, and RBCUs include maintaining containment pressure and temperature below design limits following an accident. This surveillance assures that containment heat removal capability is adequate assuming a worst case single failure. ITS SR 3.6.5.4 requires that the containment heat removal capability be verified on an 18 month frequency. Since service induced fouling can reduce containment heat removal capability, a fouling rate must be determined in order to establish a more frequent test interval if required.

REFERENCE

N/A

16.6 ENGINEERED SAFETY FEATURES

16.6.4 Low Pressure Injection (LPI) System Leakage

COMMITMENT: The maximum allowable leakage from the LPI System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.4.1	Verify leakage from the portion of the LPI System, except piping from the containment emergency sump to the low pressure injection pump suction isolation valve, that is outside the containment is within the limit either by use in normal operation or by hydrostatically testing at $\geq$ 350 psig.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%
SR 16.6.4.2	Verify leakage from piping from the containment emergency sump to the LPI pump suction isolation valve is within limit when tested at $\geq$ 59 psig.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.6.4.3	Verify leakage is within limit by visual inspection for excessive leakage from components of the system.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 4.5.5 and Technical Specification Interpretetion (TSI) 4.5.2/3.3.2 during the conversion to ITS.

Excessive leakage shall be measured by collection and weighing or by another equivalent method. The leakage rate limit for the Low Pressure Injection System is a judgement value based on assuring that the components can be expected to operate with-out mechanical failure for a period on the order of 200 days after a loss of coolant accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Low Pressure Injection System (59 psig) is equivalent to the design pressure of the containment. The dose to the thyroid calculated as a result of this leakage is 0.76 rem for a two-hour exposure at the site boundary.

SURVEILLANCE TEST PRESSURE

Leakage is measured during refueling outage surveillance testing. The surveillance testing may be performed as a hydrostatic test, or measured during system operation. Hydrostatic testing is used for the RBES suction lines, from the sump to LP-19/20. Required testing for the remaining portions of the system is accomplished through leakage measurement during alignment and/or operation of the LPI System, test pressure may be below 350 psig as necessary to meet operating limits. The test pressure of the rest of the LPI System is at or above the specified 59 psig or 350 psig.

Leakage rates corrections can be made, as described in the Bases, for the actual pressure versus the specified test pressure. These corrections are small - less than 10%.

The following practices are used to ensure that requirements are met:

a) The test pressure for the discharge side of the LPI System is established as near to 350 psig as practical, within the operating limits in plant procedures.

b) If the total measured leakage (uncorrected) is above 1.8 gph, the measured leakage will be evaluated for pressure differences prior to comparison to the 2 gph limit.

LEAKAGE OBSERVED DURING UNIT OPERATION

LPI System leakage found during operation in MODES 1, 2, 3 or 4, which would be in excess of 2 gph at the specified pressures of this SLC, requires entry into the appropriate Actions of ITS 3.5.3 or ITS LCO 3.0.3.

The affected train, based on the location of the leakage, must be declared out of service in accordance with the appropriate Actions of ITS 3.5.3. In addition, the point of leakage must be isolated. As an alternative to isolating the leakage, provisions may be made to take steps, in the event of a LOCA, which would isolate the leakage or prevent it from exceeding 2 gph during emergency sump recirculation. If the LPI train remains capable of performing its safety functions after isolating the leakage, or if the planned steps in the event of a LOCA would still allow all subsequent safety functions to be performed by each train, the Action of ITS 3.5.3 can be exited.

If leakage exists in each train, and would be greater than 2 gph from each train, ITS LCO 3.0.3 must be entered. Also, if there is a point of leakage which would exceed 2 gph, and this leakage could not be isolated or reduced to less than 2 gph during emergency sump recirculation of at least one operable LPI train, ITS LCO 3.0.3 must be entered.

The 2 gph limit is based upon assuring that there is no existing leakage which could indicate that mechanical components might not continue to operate during long term core cooling. It is also based upon limiting the amount of off site dose due to leakage of highly contaminated primary coolant that is recirculated from the RB emergency sump. The 59 psig / 350 psig pressures are conservative values for which leakage is to be evaluated in the LPI suction / discharge piping.

SURVEILLANCE TEST PRESSURE

Performance of the leakage measurement during LPI System operation is a valid test method. This method is used for required portions of the LPI System other than the lines between the RBES and LP-19/20. However, operational limits, fluid dynamics characteristics, and other considerations prevent achieving the exact 59 psig or 350 psig pressures. In the pump suction piping, the measurements are performed at > 59 psig. When an LPI pump is on, the pressure between the pump and throttle valve(s) LP-12/14 may be higher than the required 350 psig. The pressure downstream of the throttle valve(s) is approximately equal to RCS pressure.

The fact that portions of the discharge side of the LPI System cannot be pressurized to 350 psig does not invalidate the testing. The specified 350 psig test pressure is an arbitrary value for evaluation of leakage, rather than a true hydrostatic test pressure. (The lowest design pressure in this portion of the system is 470 psig. Leakage rate, not structural integrity, is being evaluated.) However, the discharge side measurements are performed with the LPI System as near as practical to its operating limit (300 psig for Unit 1 and 2, 290 psig for unit 3). Given any particular flow path, flow rate varies primarily with the square root of the pressure drop across the flow path. Therefore, the measured leakage rate can be adjusted for a difference between the actual test pressure and the specified test pressure as follows:

MEASURED LEAKAGE X SQUARE ROOT (SPECIFIED PRESS / ACTUAL PRESS)

Neglecting the adjustment is conservative where the actual pressure is higher than specified. Neglecting the adjustment is slightly non-conservative where the actual pressure is lower than the specified pressure. For example, if actual test pressure were 310 psig at a point of leakage measured as 1.0 gph, the normalized leakage for 350 psig would be about 1.06 gph. Because the correction is small, adjustment of the measured leakages is not necessary when uncorrected leakage is well within the 2 gph limit (i.e., when uncorrected leakage is < 1.8 gph).

Water temperatures are different during testing of different portions of the system, so that density corrections could be considered. Because higher temperatures result in conservative offsite dose results due to evaporation rate, the UFSAR, assumes temperatures of 252 degrees and 115 degrees on the LPI System suction and discharge sides, respectively. However, the UFSAR evaluation expresses leakage in drops/minute, and converts drops per minute to cc/hr without considering temperatures. Actual leakage measurements are made by a combination of drop counting and/or collection of leakage in a container (which approaches room temperature). Therefore, correction of measured leakage for temperature/density effects goes beyond the precision assumed in the analysis, and would conflict with accepted measurement techniques. It is concluded that density corrections are unnecessary.

LEAKAGE OBSERVED-DURING UNIT OPERATION

During operation in MODES 1, 2, 3 or 4, if it is determined that leakage exists such that the limit of this SLC would not be met, two actions must be taken. The affected LPI train(s) must be declared out of service, and provisions must be made to limit any leakage of recirculating RBES water, outside of containment, from exceeding 2 gph during an accident. It is not sufficient to simply declare an affected LPI train out of service. If both LPI trains are affected, such that it is not possible to take measures which would limit leakage to < 2 gph during recirculation, while also maintaining an operable LPI train, then ITS LCO 3.0.3 must be entered.

These requirements apply to the outside-of-containment leakage boundaries of those portions of the LPI System which would be used during emergency sump recirculation. Declaration of affected LPI train(s) as being out-of-service must be made upon determination that the limits of this SLC would be exceeded.

Actions to isolate leakage or provide steps to limit leakage to < 2 gph, should be completed promptly as dictated by ITS 3.5.3 Actions or ITS LCO 3.0.3.

If the LPI train(s) remain capable of performing all safety functions after isolating the leakage, or if the planned steps in the event of a LOCA would still allow all subsequent safety functions to be performed by each train, the affected train(s) can be declared operable after the isolation / provision of planned steps is completed.

Because recirculation would not be required for some accidents, and because an affected train might be needed prior to beginning recirculation, physically securing an affected train may not always be the best method for controlling leakage. Example 2 below illustrates this point. Situations in which more complex provisions are necessary should be addressed by a procedure.

The basis for these requirements is that-leakage above 2 gph could cause the offsite doses during an accident to exceed those which have been evaluated in the UFSAR. Because of the high concentration of fission products assumed to be present in the primary coolant after a large break LOCA significant off site dose is associated with even small amounts of primary coolant leakage outside containment.

These practices, and the examples below, represent a conservative application of this SLC during operation in MODES 1, 2, 3 or 4, when ITS LCO 3.5.3 also applies. The 2 gph limit is relatively restrictive. Therefore, the Compliance Section should be contacted when marginal rates of LPI System leakage could cause a unit shutdown.

Example 1:
During power operation with the LPI pumps not running, leakage around the shaft of the 'A' LPI pump is observed. The leakage is determined to be about 3 gph (from BWST head). To repair the leakage, the pump must be isolated.

The required action is to declare LPI Train 'A' out of service, and to enter the 72 hour Required Action A.1 for ITS 3.5.3. In addition, the A' pump must be isolated.

Example 2:
During power operation with the LPI pumps not running, leakage is observed around the packing of LP-9. The leakage is determined to be about 1 gph. It is also determined that this leakage would exceed 2 gph at 350 psig, but be less than 2 gph at 59 psig. Repairs do not require isolating LP-9.

The required action is to declare LPI Train -A' out of service, and to enter the 72 hour Required Action A.1 for ITS 3.5.3. Action must also be taken to control the leakage. Because isolating LP-9 is not necessary for repairs, it is preferable to leave the 'A' LPI train aligned for ES actuation even though it is declared out of service. To prevent leakage in excess of 2 gph in recirculation mode, verbal instructions and a turnover sheet item could be provided to turn off the 'A' LPI pump, prior to initiating the recirculation mode during an accident.

Example 3:
During power operation, the 3/8" line between LP-38 and LP-39 (PALS LINE ISOLATION VALVES) becomes disconnected. Leakage which would exceed 2 gph at 350 psig is observed coming from both directions (i.e., > 2 gph from 'A' and > 2 gph from 'B' The leakage cannot be controlled by closing LP-38 and LP-39.

The required action is to enter ITS LCO 3.0.3, and to attempt to reduce leakage.

Example 4:
Leakage is observed around LP-28.

This problem is not applicable to this SLC, because LP-28 is neither in the emergency sump recirculation flow path nor a boundary of this flow path.

<u>References:</u>

1) UFSAR Sections 6.0.3, 6.0.3.1, 6.0.3.2, 6.0.3.4, and 6.0.3.5

2) UFSAR Section 15.15.4 and 6.3.3.2.2

2) PT/1/A/0203/04 LPI System Leakage
PT/2/A/0203/04 LPI System Leakage
PT/3/A/0203/04 LPI System Leakage

3) OP/1/A/1104/04 LPI System Operation
OP/2/A/1104/04 LPI System Operation
OP/3/A/1104/04 LPI System Operation

4) PIR 3-090-0019 Leakage on 3LP-9 Exceeded Tech Specs

16.6 ENGINEERED SAFETY FEATURES

16.6.5 Core Flood Tank (CFT) Discharge Valve Breakers

COMMITMENT The breakers associated with the CFT discharge valves shall be locked open and tagged.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Be in MODE 3.	12 hours
	AND	
	A.2 Be in MODE 3 with RCS pressure ≤ 800 psig.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.6.5.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.3.3 during the conversion to ITS.

REFERENCE

N/A

16.6 ENGINEERED SAFETY FEATURES

16.6.6 Core Flooding System Test

COMMITMENT Perform required SRs.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.6.1	Verify that the check and isolation valves in the core flooding tank discharge lines operate properly during pressurization of the Reactor Coolant System.	------NOTE----- The provisions of SLC 16.2.7 do not apply. -------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 4.5.1.1.3 during the conversion to ITS.

With the reactor shut down, the valves in each core flooding line are checked for operability by reducing the Reactor Coolant System Pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

REFERENCE

N/A

16.6 ENGINEERED SAFETY FEATURES

16.6.7 Borated Water Storage Tank (BWST) Outlet Valve Control

COMMITMENT Manual valve LP-28 on the BWST discharge line shall be locked open.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Enter applicable ITS Condition for BWST inoperable.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.6.7.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.3.4.b during the conversion to ITS.

REFERENCE

N/A

16.6 ENGINEERED SAFETY FEATURES

16.6.8 Low Pressure Injection (LPI) System Valve Test Restrictions

COMMITMENT Perform specified SR.

APPLICABILITY: MODE 5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.6.8.1 Test valves LP-17 and LP-18.	After entry into MODE 5 from MODE 4 unless tested within the previous 92 days

BASES

The requirement(s) of this SLC section were relocated from CTS 4.5.1.2.1.a during the conversion to ITS. SR 16.6.8.1 is normally performed after entering MODE 5 but prior to exiting MODE 5 unless performed in the previous 92 days.

Power Operated Valves LP-17 and LP-18, are boundary valves between high pressure and low pressure design piping. As such, functional testing of these valves is performed during cold shutdown conditions when the Reactor Coolant System pressure is below the design pressure of the Low Pressure Injection System piping and the potential for over-pressurization of the low pressure system is eliminated.

REFERENCE

N/A

16.6 ENGINEERED SAFETY FEATURES

16.6.9 Containment Purge Valve Testing

COMMITMENT Perform specified SRs.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.9.1	------------------NOTE--------------------- 1. Not required to be performed until prior to going above MODE 3. 2. Not required to be performed for purge valves that have not been operated if conducted within the proceeding 184 days. 3. Perform after final closing when the purge valves have been operated. -- Perform leakage integrity tests.	After every entry into MODE 3 from MODE 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.6.9.2	Visually inspected and adjust or replace the valve seals of the purge isolation valves.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 4.4.4.1 and 4.4.4.3 during the conversion to ITS.

Leakage integrity tests of the purge supply and isolation valves are conducted in order to identify excessive degradation of the resilient seals. Excessive leakage past resilient seals is typically caused by severe environmental conditions and/or wear due to frequent use.

REFERENCE

N/A

16.6 ENGINEERED SAFETY FEATURES

16.6.10 Containment Hydrogen Recombiner System

COMMITMENT The Containment Hydrogen Recombiner System shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Hydrogen Recombiner System inoperable.	A.1 Restore to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.10.1	Verify the post-LOCA flow path by connecting and operating the Hydrogen Recombiner through its flow path. The Hydrogen Recombiner flow path shall circulate Reactor Building atmosphere at a flow > 50 SCFM.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.6.10.2	Visual inspection of the hydrogen recombiner unit.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%
SR 16.6.10.3	CHANNEL CALIBRATION of recombiner instrumentation channels.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%
SR 16.6.10.4	Operate a recombiner unit at design flow rate ± 10% and allow unit to reach recombination temperature.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 3.16.1 and 4.4.3 during the conversion to ITS.

The Containment Hydrogen Recombiner System shall be OPERABLE in MODES 1, 2, 3 and 4. The Recombiner System consists of one OPERABLE hydrogen recombiner unit available for connection to the associated flow path for each Oconee unit.

The Containment Hydrogen Recombiner System is required at approximately 7 days following a LOCA to limit hydrogen concentration to 4.0 percent by volume.

The Containment Hydrogen Recombiner System is utilized to maintain the post-accident containment atmosphere hydrogen concentration below its lower flammability limit of 4.0 percent by volume. The Containment Hydrogen

Recombiner System includes a portable hydrogen recombiner which will be moved to the affected unit following a LOCA, anchored to its foundation, and connected to piping penetrations. Also included is a portable control panel, which will be locally mounted near the recombiner, anchored to its foundation and connected to its motor control center and the recombiner.

The control panel mounted near the recombiner enables the operator to control and monitor system parameters for all functions of the recombiner system except containment isolation valve operation. The control and monitor functions include: process temperature indications, temperature control, flow indication, start/stop switch, low temperature timer and various annunciators. Therefore, the operational performance testing ensures operability.

The penetrations to and from the hydrogen recombiner are shared with the gaseous radiation monitoring pump. Since this pump is normally in operation and since there is no system isolation valve on the supply branch to the recombiner, the blind flanges are the only means of system isolation.

Therefore, these flange joints shall be leak tested after each removal and installation to ensure adequate isolation.

The hydrogen recombiner unit operational performance test should be conducted with full flow and with the heaters energized. The capability of the recombiner to achieve the required recombination temperature and flow rate is considered an adequate test of recombination efficiency. Gas inlet and outlet sampling is not required.

REFERENCE

UFSAR, Section 15.16

16.6 ENGINEERED SAFETY FEATURES

16.6.12 Additional High Pressure Injection (HPI) Requirements

COMMITMENT: Two HPI trains shall be OPERABLE.

-----NOTES-----

1. Three HPI pumps and the HPI discharge crossover valves shall be OPERABLE and the suction headers shall be cross-connected.

2. The HPI discharge headers shall be hydraulically separated.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One HPI pump inoperable. OR One or more HPI discharge crossover valve(s) inoperable. OR HPI suction headers not cross-connected.	A.1 Restore HPI pump to OPERABLE status. AND A.2 Restore HPI discharge crossover valve(s) to OPERABLE status. AND A.3 Cross-connect HPI suction headers.	72 hours 72 hours 72 hours
B. HPI discharge headers cross-connected.	B.1 Hydraulically separate the HPI dischage headers.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One HPI train incapable of being automatically actuated but capable of being manually actuated.	C.1 Restore capability to automatically actuate train.	24 hours
D. Required Action and associated Completion Time of Condition A, B or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Reduce RCS temperature to $\leq$ 350°F.	12 hours 60 hours
E. One HPI train incapable of being automatically actuated and incapable of being manually actuated.	E.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.6.12.1	Perform CHANNEL CHECK for each HPI discharge crossover valve flow instrument.	31 days
SR 16.6.12.2	Perform CHANNEL CALIBRATION for each HPI discharge crossover valve flow instrument.	18 months
SR 16.6.12.3	Cycle each HPI discharge crossover valve.	18 months

BASES

This SLC imposes operability requirements regarding the high pressure injection (HPI) system in addition to those imposed by Limiting Condition for Operation (LCO) 3.5.2, "High Pressure Injection (HPI)." These additional requirements are:

1. The third HPI pump and the HPI discharge crossover valves are required to be OPERABLE and the suction headers are required to be cross-connected when THERMAL POWER is ≤ 60% RTP;

2. The HPI discharge headers are required to be hydraulically separated whenever the plant is operating in a MODE or condition which requires the HPI system to be OPERABLE; and

3. Surveillance Requirements have been added to confirm the OPERABILITY of the HPI discharge crossover valves and their associated flow instruments.

These additional requirements are necessary, because the requirements of the Improved Technical Specifications (ITS) were based on the previous Technical Specifications (referred to as the CTS) which were deficient. Due to the time limitations associated with the completion of the review of the ITS, the NRC and Duke agreed to resolve the deficiencies regarding these requirements in another License Amendment Request.

In the following letters, Duke informed the NRC of deficiencies with the CTS requirements regarding the HPI System: 1) Licensee Event Report (LER) 50-269/90-15 dated December 20, 1990 (Reference 1), 2) License Amendment Request dated March 31, 1997 (Reference 7), 3) Special Report dated October 22, 1990 (Reference 9), and 4) License Amendment Request dated December 16, 1998. The License Amendment Request submitted on December 16, 1998 (Reference 8), resolves the deficiencies identified in the above letters.

In addition, another deficiency was discovered regarding the lack of Surveillance Requirements to demonstrate the OPERABILITY of the flow instruments associated with the HPI discharge crossover valves prior to the implementation of the ITS.

<u>Commitment</u>

Two HPI trains are required to be OPERABLE when in MODES 1 and 2, and MODE 3 with reactor coolant system (RCS) temperature > 350°F. This requirement is consistent with LCO 3.5.2.

Commitment Note 1

This Commitment Note establishes that the third HPI pump and the HPI discharge crossover valves (i.e., HP-409 and HP-410) shall be operable and the HPI suction headers shall be cross-connected whenever the plant is operating with RCS temperature > 350°F. The requirement is consistent with the Note in ITS 3.5.2 for operation with THERMAL POWER > 60% RTP, and it replaces the interpretation of CTS 3.3.1 approved on November 26, 1990 (Reference 2), regarding operation with THERMAL POWER < 60% RTP.

ITS 3.5.2 provides requirements for the HPI System when the RCS temperature is > 350°F. The ITS requirements captured the CTS requirements which were based on the analysis of a Small Break Loss of Coolant Accident (SBLOCA) which assumed a break on the discharge side of the reactor coolant pumps. The analysis concluded that one HPI train had sufficient capacity to mitigate SBLOCAs when reactor power was < 60% full power. As reported in LER 269/90-15 (Reference 1), Duke discovered that the analysis was non-conservative. It assumed:

1) an even flow split between the injection line connected to the broken cold leg and the injection line connected to the intact cold leg. The even flow split resulted from the assumption that the back pressure on each line was equal to RCS pressure; and

2) HPI flow from the injection line connected to the broken leg is injected into the reactor coolant pump discharge volume. A computer model then determined how much of the injection flow is lost out the break.

In the LER, Duke reported that one HPI train was inadequate to mitigate a break of an HPI injection line when reactor power was < 60% full power. In this case, the appropriate back pressure assumption would be containment pressure for the broken injection line, and RCS pressure for the intact injection lines. Additionally, none of the HPI flow through the broken injection line would reach the RCS. The resulting flow split from this asymmetric pressure boundary condition would cause less injection flow to reach the reactor. As a result, Duke imposed additional requirements upon the operation of Oconee Nuclear Station Units 1, 2, and 3 with reactor power < 60% full power (Reference 2). These additional requirements were equivalent to the requirements for operation with reactor power > 60% full power (i.e., a third HPI pump and HPI discharge crossover valves were required to be operable, and the HPI suction headers were required to be cross-connected).

Commitment Note 2

This Commitment Note replaces an interpretation of CTS 3.3.1.a(1) approved on January 21, 1999, which limited operation with HPI discharge headers cross-connected to 24 hours (Reference 3).

The operability of the HPI System must be maintained to ensure that no single active failure can disable both HPI trains. Additionally, while the HPI System was not designed to cope with passive failures, the HPI trains must be maintained independent to the extent possible during normal operation. The

only NRC approved exception to this principle is cross-connecting the HPI suction headers during normal operation (Reference 4). Thus, hydraulic separation of the HPI discharge headers is required during normal operation to maintain defense-in-depth (i.e., independence of the HPI discharge headers).

Actions

Required Actions A.1, A.2, and A.3

With one HPI pump inoperable, one or more HPI discharge crossover valve(s) inoperable, or the HPI suction headers not cross-connected, the HPI pump and discharge crossover valve(s) must be restored to OPERABLE status and the HPI suction headers must be cross-connected within 72 hours. The Completion Time is reasonable, because the HPI System continues to be capable of mitigating an accident. It is consistent with the Completion Times established in Action A of ITS 3.5.2, and the CTS interpretation dated November 26, 1990 (Reference 2).

Required Action B.1

In the event the HPI discharge headers are cross-connected, the HPI trains are not maintained independent to the extent possible; thus, Condition B of this SLC shall be entered. Required Action B.1 requires the HPI discharge headers to be hydraulically separated within 24 hours. It is consistent with the Completion Time established in the CTS interpretation dated January 21, 1999.

Required Action C.1

As stated in the discussion of Commitment Note 1, ITS 3.5.2 is based on the CTS requirement which utilized a non-conservative SBLOCA analysis which determined that only one HPI train was required to mitigate accidents when THERMAL POWER was < 60% RTP. As identified in References 1 and 2, both HPI trains are required to mitigate the consequences of an accident regardless of the THERMAL POWER at which the plant is operating. The Completion Time is reasonable, because the HPI system continues to be capable of mitigating an accident. It is consistent with the Completion Time established in the CTS interpretation dated November 21, 1990 (Reference 5).

In an SER dated December 13, 1978, (Reference 4) the NRC approved the use of manual operator action to place a second HPI train in operation within 10 minutes of accident initiation. Originally, this manual action was only required when THERMAL POWER was > 60% RTP. An interpretation to CTS 3.3.1(c)(2) dated November 21, 1990, limits operation with one HPI train incapable of automatic actuation but capable of being manually aligned to 24 hours with THERMAL POWER > 60% full power (Reference 5). This CTS interpretation was incorporated into Condition D of ITS 3.5.2.

Action C of this SLC is consistent with Required Action D.1 of ITS 3.5.2. This is appropriate, because: 1) the requirements with operation with THERMAL POWER > 60% RTP have not changed; and 2) the interpretation of CTS 3.3.1 approved on November 26, 1990 (Reference 2) expanded the applicability of CTS 3.3.1.c(1) to operation at < 60% full power. In the event an HPI train is

incapable of automatic actuation but capable of manual actuation, Required Action C.1 of this SLC requires the restoration of the ability to automatically actuate the HPI train within 24 hours.

Required Actions D.1 and D.2

In the event a Required Action and associated Completion Time of Condition A, B, or C is not met, the plant must be placed in a condition for which the SLC does not apply. Required Actions D.1 and D.2 require the plant to be placed in MODE 3 within 12 hours and RCS temperature reduced to ≤ 350°F within 60 hours. These Completion Times are consistent with those provided in ITS 3.5.2, Condition E.

Required Action E.1

In the event one HPI train is incapable of being automatically actuated and incapable of being manually actuated, the plant is required to enter LCO 3.0.3 immediately. In this condition, the HPI System is unable to perform its safety function. The immediate Completion Time is appropriate given the safety significance of the condition.

Surveillance Requirements

SR 16.6.12.1 and SR 16.6.12.2

This SLC specifies Surveillance Requirements for the flow instruments associated with the HPI discharge crossover valves. SLC SR 16.6.12.1 and SR 16.6.12.2 require a CHANNEL CHECK and CHANNEL CALIBRATION be performed for these flow instruments at a Frequency of 31 days and 18 months, respectively. If one or both of these SRs is not met, the associated HPI discharge crossover valve (i.e., HP-409 or HP-410) must be declared inoperable, because the Bases for Technical Specification 3.5.2 requires that the associated flow instrument be OPERABLE to support the valve's OPERABILITY.

SR 16.6.12.3

Periodic stroke testing of the HPI discharge crossover valves (i.e., HP-409 and HP-410) is required to ensure that the valves can be manually cycled from the Control Room. This test is performed on an 18-month Frequency.

References

1. Licensee Event Report 269/90-15, dated December 20, 1990.

2. Interpretation of CTS 3.3.1 approved on November 26, 1990.

3. Interpretation of CTS 3.3.1.a(1) approved on January 21, 1999.

4. Letter from R. W. Reid (NRC) to W. O. Parker (Duke), NRC Safety Evaluation Report on the Oconee modification adding HP-409 & HP-410, dated December 13, 1978.

5. Interpretation of CTS 3.3.1 approved on November 21, 1990.

6. UFSAR Sections 5.4.7.2, 6.3.1, 6.3.2.2.1, and 9.3.2, and Chapter 15.

7. Letter from J. W. Hampton (Duke) to U. S. NRC, License Amendment Request regarding CTS requirements for the HPI System, dated March 31, 1997.

8. Letter from W. R. McCollum (Duke) to U. S. NRC, License Amendment Request regarding CTS requirements for the HPI System, dated December 16, 1998.

9. Letter from H. B. Barron (Duke) to U. S. NRC, "Special Report Concerning High Pressure Injection Train Rendered Inoperable Due To Inappropriate Operator Actions," dated October 22, 1990.

10. ITS 3.3.8 and 3.5.2.

16.7 INSTRUMENTATION

16.7.1 Accident Monitoring Instrumentation - Noble Gas Effluent Monitor (RIA-56)

COMMITMENT The noble gas effluent monitor shall be OPERABLE.

APPLICABILITY: MODES 1 AND 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Noble Gas Effluent Monitor (RIA-56) inoperable.	A.1 Institute alternative noble gas monitoring program.	72 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.7.1.1 Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 16.7.1.2 Perform CHANNEL CALIBRATION.	18 months.

BASES

The Noble Gas Effluent Monitor (RIA-56) is utilized for detection of significant releases and release assessment.

The Alternative methods for monitoring noble gas effluent during inoperability of RIA-56 shall include one or more of the following methods:

- RIA-45 normal range noble gas monitor on the unit vent.
- RIA-46 high range noble gas monitor on the unit vent.
- Actual vent sample.
- Direct radiation readings on RIA-45 and RIA-46 sample line.

BASES (continued)

REFERENCES:

1. Generic Letter 83-37
2. Regulatory Guide 1.97, Rev. 2

16.7 INSTRUMENTATION

16.7.2 Anticipated Transients Without Scram

COMMITMENT The ATWS Mitigation Systems Actuation Circuitry (AMSAC) and Diverse Scram System (DSS) shall be OPERABLE.

APPLICABILITY: MODE 1,
MODE 2 when Keff ≥ 1.0

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both channels of AMSAC inoperable.	A.1	Restore AMSAC to OPERABLE status.	7 days
B. One or both channels of DSS inoperable.	B.1	Restore DSS to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	-------------NOTE------------ When initiated the Required Action must be completed. ----------------------------		
	C.1	Submit a written report to the NRC outlining the cause of the channel(s) or system(s) malfunction and the plans for restoring the channel(s) or system(s) to OPERABLE status.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.2.1	Perform a Channel Logic Test of AMSAC.	184 days
SR 16.7.2.2	Perform a Channel Logic Test of DSS.	184 days
SR 16.7.2.3	Perform an Actuation Test of AMSAC.	18 months
SR 16.7.2.4	Perform an Actuation Test of DSS.	18 months.

BASES

The AMSAC and DSS are provided to mitigate the consequences of anticipated transient without scram. These anticipated transients are beyond the design basis for the plant. These events are associated with a failure of the reactor to normally trip when required as defined in the references below.

The AMSAC/DSS consists of two channels and uses a two-out-of-two coincidence logic to actuate. Each channel has an AMSAC portion and a DSS portion.

The AMSAC circuitry of each channel receives input signals on low Feedwater pump Turbine (FDWPT) control oil pressure or low Feedwater pump (FDWP) discharge pressure. Upon a valid input signal to the AMSAC portions of the two AMSAC/DSS channels, an output is generated to trip the Main Turbine and start all operable Emergency Feedwater Pumps.

The DSS circuitry of each channel receives an input signal from the Inadequate Core Cooling Monitoring System RCS pressure signals. Upon a valid signal (RCS Pressure Very High / $\geq$ 2450 psig) to both DSS portions of each channel an output is generated to interrupt power to the Control Rod Drive System gate drives for regulating rod groups 5 through 7 and the auxiliary gate drives.

An AMSAC/DSS channel is considered operable if it has met the surveillance criteria of this commitment and the AMSAC/DSS enabled light (located in the control room) is on <u>and</u> the AMSAC/DSS Ch. 1 and Ch. 2 bypassed lights (also located in the control room) are not on <u>and</u> "Sy Max" Programmable Controllers RUN Lights (ON) <u>and</u> HALT Lights (OFF) for AMSAC/DSS Ch. 1 <u>AND</u> AMSAC/DSS Ch. 2.

An Actuation Test consists of a complete test from input sensors through output actuation relays.

REFERENCES:

1. Code of Federal Regulations, Section 10 CFR 50.62 "The ATWS Rule".
2. B&WOG Generic ATWS Design Basis Document 47-1159091-00, October 9, 1985.
3. NRC Safety Evaluation Report on 47-1159091, June 30, 1988.
4. AMSAC and DSS Final Design Description, August 30, 1988.
5. NRC Safety Evaluation Report for Final Design of Oconee ATWS Modification (TACS 59119/59120/59121), November 29, 1989.

16.7 INSTRUMENTATION

16.7.3 Emergency Feedwater - Low Level Initiation

COMMITMENT Automatic low level initiation of both MDEFW pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both channels inoperable.	A.1 Restore to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	-------------NOTE------------ When initiated the Required Action must be completed. -----------------------------	
	B.1 Submit a written report to the NRC outlining the cause of the channel(s) or system(s) malfunction and the plans for restoring the channel(s) or system(s) to OPERABLE status.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.3.1	Perform a CHANNEL CALIBRATION.	18 months

BASES

The Steam Generator Level Control System (SGLCS) receives four OTSG level signals. Each train receives one signal from each OTSG (OTSG A & B Level to Train A and the same to Train B). These level signals are used to start the MDEFWPs upon 2 out of 2 low level in either OTSG. A level signal indicating below the initiation setpoint or failed low is considered to be operable.

The most limiting transient for the EFW system is the Loss of Main Feedwater (LMFW), (Ref. UFSAR Section 10.4.7). The primary success path to mitigate the LMFW includes initiation of the EFW system. The UFSAR evaluation credits automatic initiation of EFW on loss of both main feedwater pumps as sensed by low hydraulic oil pressure. In addition, for plant conditions in which automatic initiation circuitry must be disabled (i.e., turbine header pressure < 850 psig) adequate time is available for manual initiation of EFW. Thus, initiation of EFW on low OTSG level is not credited for any DBA or transient. EFW initiation on low OTSG level has been included as a SLC in response to GL 89-19 and USI A-47 and provides additional protection from OTSG dryout.

EFW initiation on low OTSG level is applicable above 250°F, although it is not required for operability of the EFW System.

In order to provide additional protection from OTSG dryout, RCS temperature may not be increased above 250°F with low level initiation of MDEFW inoperable. However, if the Unit is above 250°F, shutdown is not required since low level initiation is not credited for any DBA or transient.

REFERENCES:

1. Generic Letter 89-19, Safety Implication of Control Systems

16.7 INSTRUMENTATION

16.7.4 -----DELETED-----

16.7 INSTRUMENTATION

16.7.5 Steam Generator Overfill Protection

COMMITMENT The steam generator overfill protection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 when RCS T_{ave} > 325°F

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Steam generator overfill protection system inoperable.	A.1 Restore to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	-------------NOTE------------ When initiated the Required Action must be completed. ---------------------------- B.1 Submit a written report to the NRC outlining the cause of the channel(s) or system(s) malfunction and the plans for restoring the channel(s) or system(s) to OPERABLE status.	 30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.5.1	Perform Trip Test (SV6).	92 days
SR 16.7.5.2	Perform Trip Test (SV12).	18 months
SR 16.7.5.3	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 16.7.5.4	Perform CHANNEL CALIBRATION.	18 months.

BASES

BACKGROUND

This commitment supports closure of Generic Letter 89-19 by providing limiting conditions for operation, actions, and surveillances for steam generator overfill protection.

APPLICABLE SAFETY ANALYSES

Steam generator overfill protection (e.g.. the high steam generator level feedwater pump trip) plays an important role in the mitigation of Main Feedwater (MFW) overfill events that could lead to Pressurized Thermal Shock (PTS). Studies have been performed by Duke, the B&W Owners Group, and Oak Ridge National Laboratory (ORNL) to assess the probability of vessel failure due to PTS events. The high level trip is credited in many of these studies to mitigate overfill transients; thus, the PTS results are highly dependent on the functioning of the high level trip. However, the PTS sequences which lead to core melt contribute less than 1% to the overall calculated core melt frequency.

APPLICABILITY

The overfill protection system is required to be operable for RCS temperatures above 325°F to assure that an overcooling event due to steam generator

overfill will not lead to pressurized thermal shock of the reactor vessel. For RCS temperatures ≤ 325° F, The Low Temperature Overpressure Protection (LTOP) system provides protection against overpressure concerns.

COMMITMENT

Steam generator overfill protection is provided through the ICS to terminate main feedwater when the high level setpoint is reached. Two transmitters per steam generator monitor steam generator water level. Protection is provided by 2 out of 2 logic on either steam generator which actuates two trip devices. The high level monitoring circuits deenergize to trip: thus a deenergized module is operable. Two trip devices (SV6 and SV12) are provided on each MFWPT. For example, 2 out of 2 logic on the "A" steam generator will actuate both trip devices-on both MFWPTs. Since either steam generator can cause an overcooling event, then the overfill protection logic for both steam generators are required to be operable for the overfill protection system to be considered operable.

ACTIONS

The 72 hour completion time in Required Action A.1 provides an adequate level of availability of the overfill protection system for performing its function while allowing reasonable time to permit necessary maintenance on the system.

SURVEILLANCE REQUIREMENTS

SR 16.7.5.1

This surveillance verifies that the SV6 trip device will trip the associated MFWPT. SV6 is exercised by the "oil trip" test. When the oil trip is exercised, SV6 is energized thus tripping the overspeed governor which trips the mechanical trip mechanism of the MFWPT. This surveillance can be performed on line and is part of the secondary system protection test. The 92 day frequency for this Surveillance was determined to be adequate based on operating experience.

SR 16.7.5.2

This surveillance verifies that the SV12 trip device will trip the associated MFWPT. This Surveillance can only be performed when the MFWPT is out of service. The 18 month frequency for this Surveillance was determined to be adequate based on operating experience.

SR 16.7.5.3

This surveillance requires a CHANNEL FUNCTIONAL TEST which verifies a trip signal is provided in response to high steam generator level. The 18 month frequency for this Surveillance was determined to be adequate based on operating experience.

SR 16.7.5.4

This surveillance requires a CHANNEL CALIBRATION which verifies the channel responds to steam generator level with the necessary range and accuracy. This surveillance is also required by ITS SR 3.3.8-1 for Item 12. The 18 month frequency for this surveillance was determined to be adequate based on operating experience.

REFERENCES:

1. Generic Letter 89-19, Request for Action Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants."
2. H. B. Tucker (Duke) to NRC Document Control Desk, Response to GL 89-19, March 19, 1990

16.7 INSTRUMENTATION

16.7.6 -----DELETED-----

16.7 INSTRUMENTATION

16.7.7 Position Indicator Channel Testing

COMMITMENT Perform specified SR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.7.1	Perform CHANNEL CALIBRATION of the Absolute and Relative Position Indication Channels for each CONTROL ROD.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS Table 4.1-1, Item 23 and 24 during the conversion to ITS.

Calibration of the CONTROL ROD Absolute and Relative position indication channels supports OPERABILITY of the CONTROL ROD position indication channels required by ITS LCO 3.1.7

REFERENCES

1. ITS B 3.1.7

2. UFSAR, Section 7.6

16.7 INSTRUMENTATION

16.7.8 Incore Instrumentation

COMMITMENT Incore detectors shall be OPERABLE as follows:

a. At least three detectors in each of at least three strings shall lie in the same axial plane, with one plane in each axial core half. The axial planes in each core half shall be symmetrical about the core mid-plane. The detector strings shall not have radial symmetry.

b. At least two sets of at least four detectors shall lie in each axial core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane. Detectors in the same plane shall have quarter core radial symmetry.

APPLICABILITY: MODE 1 with ALLOWABLE THERMAL POWER $\geq$ 80% RTP

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Incore detectors inoperable.	A.1	Discontinue using incore detectors to determine axial imbalance or quadrant power tilt.	Immediately
	AND		
	A.2	Reduce ALLOWABLE THERMAL POWER < 80% RTP.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.8.1	Check functioning of incore detectors, including a check of the computer readout or recorder readout.	31 days

BASES

The requirement(s) of this SLC section were relocated from CTS 3.5.4 and Table 4.1-1, Item 34 during the conversion to ITS.

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

The safety of reactor operation at or below 80 percent of the power allowable for the reactor coolant pump combination (Ref. 1) without the axial imbalance trip system has been determined by extensive 3-D calculations, and was verified during the physics startup testing program.

REFERENCES

1. UFSAR, Section 5.1.2.3

16.7 INSTRUMENTATION

16.7.9 RCP Monitor

COMMITMENT Four RCP monitor channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place channel in trip.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.7.9.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS Table 3.5.1-1, Note h during the conversion to ITS.

The RCP monitors provide inputs to the RCP pump to power trip channel logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

REFERENCES

UFSAR, Section 7.2

16.7 INSTRUMENTATION

16.7.10 Core Flood Tank (CFT) Instrumentation

COMMITMENT One level instrument channel and one pressure instrument channel shall be OPERABLE for each CFT.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Be in MODE 3.	12 hours
	AND	
	A.2 BE in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.7.10.1 Perform CHANNEL CHECK.	12 hours
SR 16.7.10.2 Perform CHANNEL CALIBRATION.	------NOTE----- The provisions of SLC 16.2.7 do not apply. -------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 3.3.3 and Table 4.1-1, Item 25 during the conversion to ITS.

REFERENCES

N/A

16.7 INSTRUMENTATION

16.7.11 Display Instrumentation

COMMITMENT Perform specified Surveillance Requirements for each Function in Table 16.7.11-1.

APPLICABILITY: According to Table 16.7.11-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.11.1	Perform CHANNEL CHECK.	12 hours
SR 16.7.11.2	Perform CHANNEL CHECK.	24 hours
SR 16.7.11.3	Perform CHANNEL CHECK.	31 days
SR 16.7.11.4	Perform battery check.	31 days
SR 16.7.11.5	Perform functional test.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.7.11.6	Perform CHANNEL CALIBRATION.	12 months
SR 16.7.11.7	Perform CHANNEL CALIBRATION.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from TCTS Table 4.1-1, Items 22, 27, 31, 32, 33, 35, 36, 38, 40, and 50 during the conversion to ITS.

REFERENCES

N/A

Table 16.7.11-1 (page 1 of 1)
Display Instrumentation

Function	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS
1. Pressurizer temperature	1, 2, 3	SR 16.7.11.1 SR 16.7.11.7
2. Letdown storage tank level	1, 2, 3	SR 16.7.11.2 SR 16.7.11.7
3. Boric Acid Mix Tank Level	1, 2, 3	SR 16.7.11.6
4. Boric Acid Mix Tank Temperature	1, 2, 3	SR 16.7.11.3 SR 16.7.11.6
5. CBAST Level	1, 2, 3	SR 16.7.11.6
6. CBAST Temperature	1, 2, 3	SR 16.7.11.3 SR 16.7.11.6
7. Containment Temperature	1, 2, 3	SR 16.7.11.7
8. Emergency Plant Radiation Instruments	At all times	SR 16.7.11.4 SR 16.7.11.7
9. Environmental Monitors	At all times	SR 16.7.11.5 SR 16.7.11.7
10. Reactor Building Emergency Sump Level	1, 2, 3	SR 16.7.11.7
11. Turbine Overspeed Trip	1, 2, 3	SR 16.7.11.7
12. PORV Position	1, 2, 3	SR 16.7.11.3 SR 16.7.11.7
13. Primary System Safety Relief Valve Position	1, 2, 3	SR 16.7.11.3 SR 16.7.11.7

16.7 INSTRUMENTATION

16.7.12 SSF Diesel Generator (DG) Air Start System Pressure Instrumentation

COMMITMENT One SSF DG Air Start System Pressure instrument channel shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One instrument channel inoperable.	A.1 Restore instrument channel to OPERABLE status.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.7.12.1 Perform CHANNEL CHECK.	7 days
SR 16.7.12.2 Perform CHANNEL CALIBRATION.	12 months

BASES

The requirement(s) of this SLC section were relocated from CTS Table 3.18-1, Item 6 and Table 4.20-1, Item 9 during the conversion to ITS.

The surveillance requirements for the SSF Instrumentation are based on experience in operation of both conventional and nuclear systems. The minimum checking frequency stated is deemed adequate for SSF Instrumentation. Calibration is performed to assure the presentation and acquisition of

accurate information. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

REFERENCES

UFSAR Section 9.6.1.

16.7 INSTRUMENTATION

16.7.13 SSF Instrumentation

COMMITMENT Perform specified Surveillance Requirements for each Function in Table 16.7.13-1.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.7.13.1	------------------NOTE------------------- This Surveillance shall be performed when the associated pump is operated during IST. -- Perform CHANNEL CHECK.	92 days
SR 16.7.13.2	Perform CHANNEL CALIBRATION.	12 months
SR 16.7.13.3	Perform CHANNEL CALIBRATION.	------NOTE----- The provisions of SLC 16.2.7 do not apply. -------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from Technical Specification Table 4.20-1, Items 2, 5, 7 and 8 during the conversion to Improved Technical Specifications.

The surveillance requirements for the SSF Instrumentation are based on experience in operation of both conventional and nuclear systems. The minimum checking frequency stated is deemed adequate for SSF Instrumentation. Calibration is performed to assure the presentation and acquisition of accurate information. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

REFERENCES

N/A

Table 16.7.13-1 (page 1 of 1)
SSF Instrumentation

Function	SURVEILLANCE REQUIREMENTS
1. SSF RC Makeup Pump Suction Pressure	SR 16.7.13.1 SR 16.7.13.3
2. SSF RC Makeup Pump Discharge Pressure	SR 16.7.13.1 SR 16.7.13.3
3. SSF RC Makeup Pump Suction Temperature	SR 16.7.13.1 SR 16.7.13.3
4. SSF RC Makeup Pump Discharge Flow	SR 16.7.13.1 SR 16.7.13.3
5. SSF Auxiliary Service Water Pump Suction Pressure	SR 16.7.13.1 SR 16.7.13.2
6. SSF Auxiliary Service Water Pump Discharge Pressure	SR 16.7.13.1 SR 16.7.13.2
7. SSF Auxiliary Service Water Pump Unit 1 Flow	SR 16.7.13.2
8. SSF Auxiliary Service Water Pump Unit 2 Flow	SR 16.7.13.2
9. SSF Auxiliary Service Water Pump Unit 3 Flow	SR 16.7.13.2
10. SSF Auxiliary Service Water Pump Discharge Test Flow	SR 16.7.13.1 SR 16.7.13.2
11. SSF Auxiliary Service Water Pump Suction Temperature	SR 16.7.13.1 SR 16.7.13.2
12. Underground Fuel Oil Storage Tank Inventory	SR 16.7.13.2
13. D/G Service Water Pump Discharge Flow	SR 16.7.13.1 SR 16.7.13.2
14. D/G Service Water Pump Discharge Pressure	SR 16.7.13.1 SR 16.7.13.2

16.8 ELECTRIC POWER SYSTEM

16.8.1 Control of Room Temperature for Station Blackout

COMMITMENT Control Room, Cable Room and Electrical Equipment Room temperatures shall be within limits.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more room temperatures exceeding limit.	A.1 Initiate action to restore temperature to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.1.1	Verify Control Room temperature is nominally ≤ 85°F.	24 hours
SR 16.8.1.2	Verify Cable Room temperature is nominally ≤ 85°F.	24 hours
SR 16.8.1.3	Verify Electrical Equipment Room temperature is nominally ≤ 90°F.	24 hours

BASES

BACKGROUND

This commitment provides guidance on the need to ensure that the initial room temperatures assumed in the room heat up calculations for the Oconee's response to the Station Blackout Rule, 10 CFR 50.63, are maintained. The requirements of the Station Blackout Rule are that the control rooms not exceed 120°F during the entire coping duration of a Station Blackout. The requirements for the other rooms are that they not exceed temperatures for which the equipment contained in them would not reliably operate. It has been shown by room heat up calculations that Oconee can meet the temperature requirements assuming initial starting temperatures are not greater than specified above which are greater than normal operating temperature for these rooms.

If these room temperatures are not maintained at the normal operating value below those specified, then the ability to cope with a Station Blackout for the entire 4 hour duration would be affected. The room temperatures could exceed the assumed limits prior to the end of the 4 hours. When normal room cooling cannot maintain normal room temperatures, then immediate action should be taken to supplement room cooling or provide a backup cooling method. This will reduce the vulnerability to exceeding the upper temperature limits assumed in the event of a Station Blackout.

APPLICABLE SAFETY ANALYSES

Maintaining normal room temperatures assures that in the event of a Station Blackout, Operators can remain in the control rooms for the duration of the required 4 hour coping time period. Additionally, equipment located in the cable rooms and electrical equipment rooms will continue to function throughout the 4 hour period. This commitment has been identified to the NRC in a letter dated July 1, 1992, "Revised Response to 10 CFR 50.63, Requirements for Station Blackout."

REFERENCES:

1. 6/12/92 letter from the NRC to Duke Power Company, Oconee Nuclear Station Units 1, 2, and 3, "Summary of June 4, 1992, Meeting on Station Blackout Response for the Oconee Nuclear Station."
2. 7/01/92 letter from J. W. Hampton to the NRC, "Revised Response to 10 CFR 50.63 Requirements for Station Blackout."
3. 10 CFR 50.63, Loss of All Alternating Current Power.
4. OSC 4747, Attachment 7, Rev. 3.
5. PIP 0-094-0668.

16.8 ELECTRIC POWER SYSTEM

16.8.2 -----DELETED-----

16.8 ELECTRIC POWER SYSTEM

16.8.3 Power Battery Parameters

COMMITMENT Power Battery parameters shall be within specified limits.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Electrolyte level below top of cell plates. <u>OR</u> Battery cell float voltage < 2.06 volts. <u>OR</u> Electrolyte Temperature < 60°F. <u>OR</u> No battery chargers are available to a battery.	A.1 Declare associated battery inoperable.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. A single battery inoperable.	B.1	Declare associated distribution center (1DP, 2DP, 3DP) inoperable.	Immediately
	<u>AND</u>		
	B.2	--------NOTE--------- Not required when associated buses (PA or PB) are cross-tied for all ONS units. --------------------- Declare Turbine Driven Emergency Feedwater (TDEFW) System and Anticipated Transients Without Scram (ATWS) System inoperable.	Immediately
	<u>AND</u>		
	B.3	Initiate action to cross-tie the associated buses (PA or PB) for all ONS Units.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more batteries inoperable.	C.1	Declare associated distribution center (1DP, 2DP, 3DP) inoperable.	Immediately
	<u>AND</u>		
	C.2	Declare Turbine Driven Emergency Feedwater (TDEFW) System and Anticipated Transients Without Scram (ATWS) System inoperable.	Immediately
D. One battery charger inoperable.	D.1	Initiate action to connect the standby charger to the associated bus.	Immediately
E. Electrolyte level < minimum or > maximum level indication marks.	E.1	Restore electrolyte level to within limits.	90 days
F. Battery cell float voltage < 2.13 Volts and ≥ 2.06 Volts.	F.1	Restore cell float voltage to within limits.	90 days
G. Required Action and associated Completion Time not met.	G.1	Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.3.1	Verify pilot cell float voltage ≥ 2.13 VDC.	7 days
SR 16.8.3.2	Verify pilot cell electrolyte level > minimum and < maximum level indication marks.	7 days
SR 16.8.3.3	Verify each cell float voltage ≥ 2.13 VDC.	92 days
SR 16.8.3.4	Verify each cell electrolyte level > minimum and < maximum level indication marks.	92 days
SR 16.8.3.5	Verify temperature of every sixth connected cell > 60°F.	92 days

BASES

BACKGROUND

This SLC on the 250VDC Power Battery Cell Parameters utilizes the limits on electrolyte level, float voltage, and temperature for the 250VDC Power Batteries to determine operability of the batteries. Float voltage is the voltage that is required to be continuously applied to the battery which is sufficient to maintain a constant state of charge. The limits for the designated pilot cell's float voltage, electrolyte level, and temperature is characteristic of a charged cell with adequate capacity. The limits for each connected cell's float voltage, electrolyte level and temperature ensures the OPERABILITY and capability of the battery.

In addition, the SLC provides the required actions for restoring the system to an OPERABLE status should a battery or charger become inoperable.

APPLICABLE SAFETY ANALYSIS

The 250VDC Power Batteries provide DC power for the Turbine Driven Emergency Feedwater (TDEFW) System. The OPERABILITY of the 250VDC Power Batteries is required to ensure the operability and the capability of the TDEFW system. The TDEFW system is required to be operable in accordance with ITS 3.7.5 In addition, the Anticipated Trip Without Scram (ATWS) system is supported by the power batteries. Selected Licensee Commitment 16.7.2 provides the operability requirements of the ATWS system. In order to maintain the required 250VDC Power Batteries OPERABLE, Battery Cell Parameters must be maintained within specific limits.

APPLICABILITY

The Power Battery Cell Parameters are required to be within limits when the associated DC sources are required to be OPERABLE.

ACTIONS

The pilot cells are monitored closely as a measure of battery performance. Because pilot cells lose more electrolyte than the other cells, the designation of the pilot cell should be rotated among all cells in the battery. The Completion Times are based on engineering judgment considering operating experience, and the time required to complete the Required Actions.

A.1

If the electrolyte level is below the top of the cell plates, the entire battery is conservatively assumed to be inoperable, because the cell's discharge capacity would be reduced, and the plates may suffer permanent damage. The battery may be restored to OPERABLE status by restoring the electrolyte level in accordance with the Required Actions of the SLC.

If the float voltage of a battery cell is < 2.06 volts, the battery is assumed to be inoperable, because battery voltage may not be adequate to carry

required loads. The battery may be restored to OPERABLE status by restoring the float voltage to ≥ 2.06 volts in accordance with the Required Actions of the SLC.

If the electrolyte temperature of a connected cell is < 60°F, the associated battery must be declared inoperable and the Required Actions taken as appropriate. With temperature < 60°F, the battery's capability may not be sufficient to meet the design basis load demand.

If no battery charger is available to a battery, then the associated battery shall be declared inoperable. The associated DC buses on all ONS units can be cross-tied to ensure operability of the system.

B.1, B.2. and B.3

If a single battery is inoperable, then the associated DC buses (PA or PB) on all ONS units can be cross-tied to ensure operability of the system. The TDEFW system and ATWS are considered operable in this configuration. If the DC buses are not cross-tied then the associated distribution center (1DP, 2DP, 3DP) is inoperable. The TDEFW system and ATWS on the associated unit are NOT considered operable in this configuration.

C.1 and C.2

If two or more batteries are inoperable, then the associated distribution centers (1DP. 2DP, 3DP) are inoperable. The TDEFW system and ATWS on the associated units are NOT considered operable in this configuration. Inadequate battery capacity is available to operate the PA or PB buses cross-tied with two PA or two PB batteries unavailable. In addition, excessive fault current (greater than protective device ratings) is available with a PA and PB battery unavailable and both PA and PB buses cross-tied.

D.1

If a battery charger is inoperable, then the Standby Charger can be connected to the associated DC bus to ensure operability of the system. The TDEFW system and ATWS are considered operable in this configuration.

E.1

The limits on electrolyte level ensures no physical damage to the plates occurs and adequate electron transfer capability is maintained.

F.1

A float voltage limit of greater than or equal to 2.13 volts will ensure the cell remains fully charged with adequate capacity.

G.1

If the appropriate parameters cannot be restored in accordance with the Required Actions, the associated battery is assumed to be inoperable.

SURVEILLANCE REQUIREMENTS

SR 16.8.3.1

This Surveillance is consistent with the recommendations of Reference 1. The reference indicates that the battery be demonstrated to meet limits on a regularly scheduled interval.

SR 16.8.3.2

This Surveillance is consistent with the recommendations of Reference 1. An adequate electrolyte level ensures that there will be a proper conductivity and capacity of the battery cell.

SR 16.8.3.3

This Surveillance is consistent with the recommendations of Reference 1. A minimum voltage is established to ensure adequate voltage to maintain cells in a constant state of charge.

SR 16.8.3.4

This Surveillance is consistent with the recommendations of Reference 1 and the battery manufacturers. An adequate electrolyte level ensures that there will be a proper conductivity path and capacity of the battery cell.

SR 16.8.3.5

This Surveillance is consistent with the recommendations of Reference 1. The electrolyte must be maintained above a minimum temperature for the battery to deliver designed power.

REFERENCES:

1. A IEEE Standard 450-1975, Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.
2. ITS 3.7.5, Emergency Feedwater System.
3. Selected Licensee Commitment 16.7.2, Anticipated Transient Without Scram.

16.8 ELECTRIC POWER SYSTEM

16.8.4 Keowee Operational Restrictions

COMMITMENT Keowee station output, and the combination of Keowee Lake Level and Operating Tailrace level shall be within the limits specified in the applicable Figures 16.8.4-1, 16.8.4-2, 16.8.4-3 and 16.8.4-4.

APPLICABILITY: Modes 1, 2, 3 and 4, during periods of commercial power generation by one or both Keowee Hydro Units (KHUs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Keowee station output not within limit. OR Combination of Keowee lake Level and operating tailrace level not within limits.	A.1 Enter applicable ITS Condition(s) and Required Actions for inoperable KHU(s), AND A.2 Initiate action to restore within limits.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.4.1	Verify combination of Keowee station output, Keowee lake level and operating tailrace level is within limits for the acceptable operating area specified in the applicable Figures 16.8.4-1, 16.8.4-2, 16.8.4-3 and 16.8.4-4.	During commercial generation

Figure 16.8.4-1
Keowee Operational Restrictions

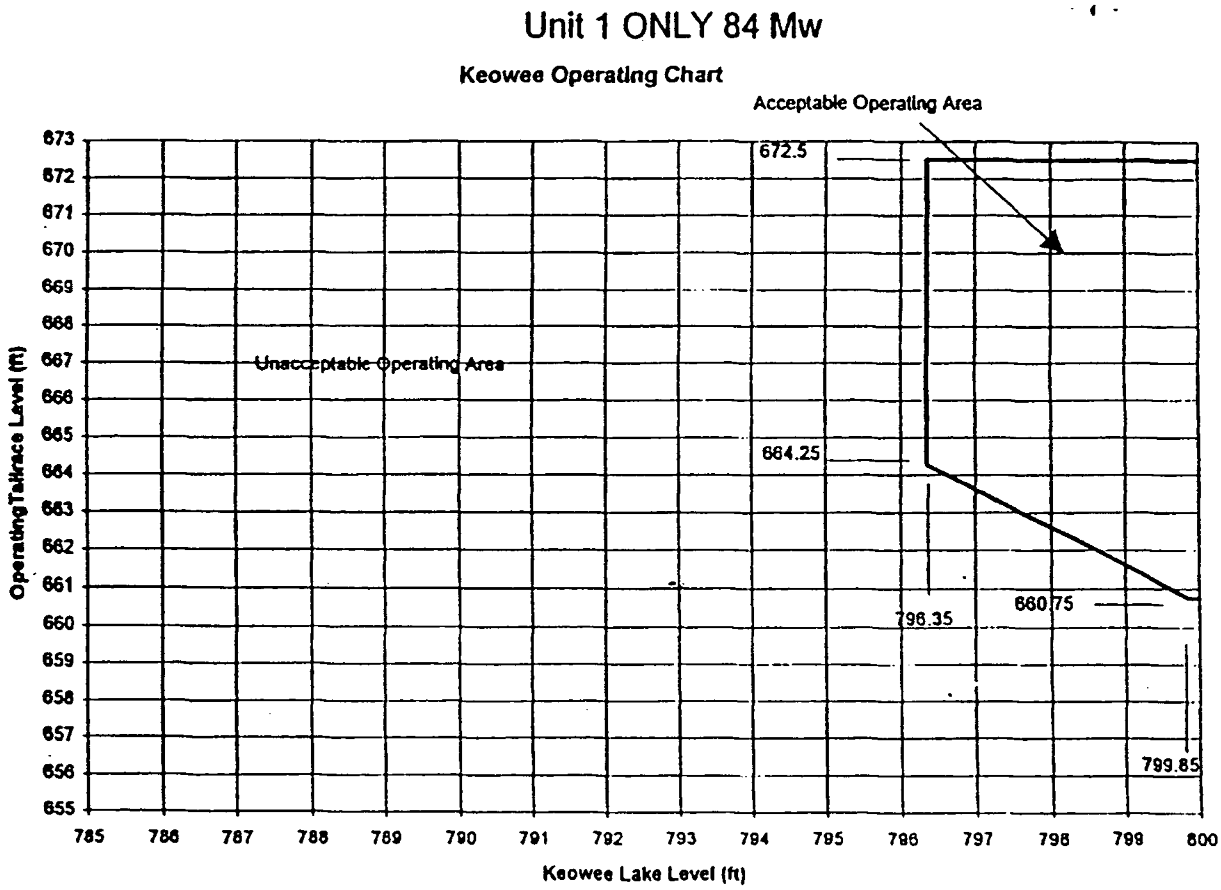


Figure 16.8.4-2
Keowee Operational Restrictions

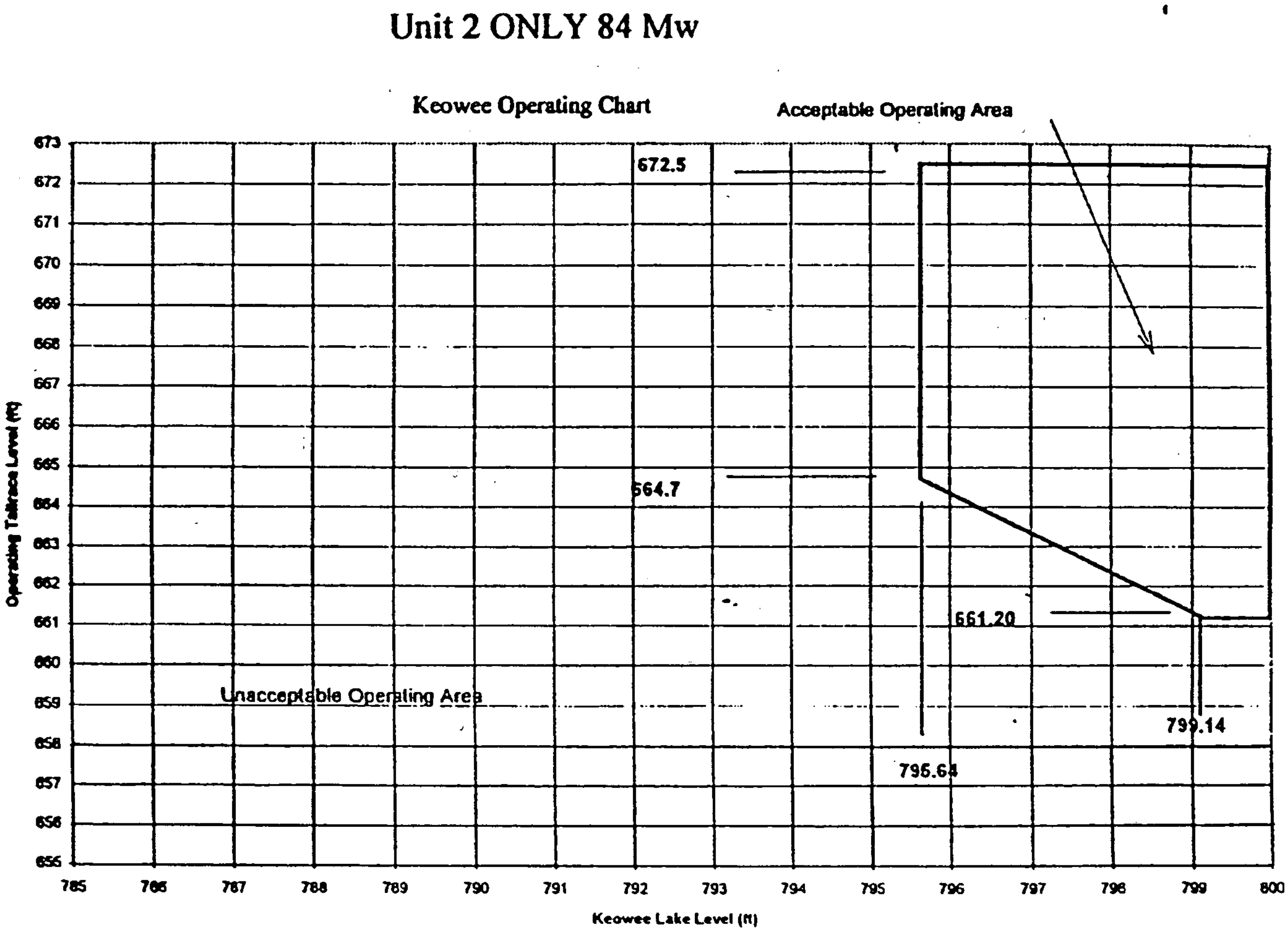


Figure 16.8.4-3
Keowee Operational Restrictions

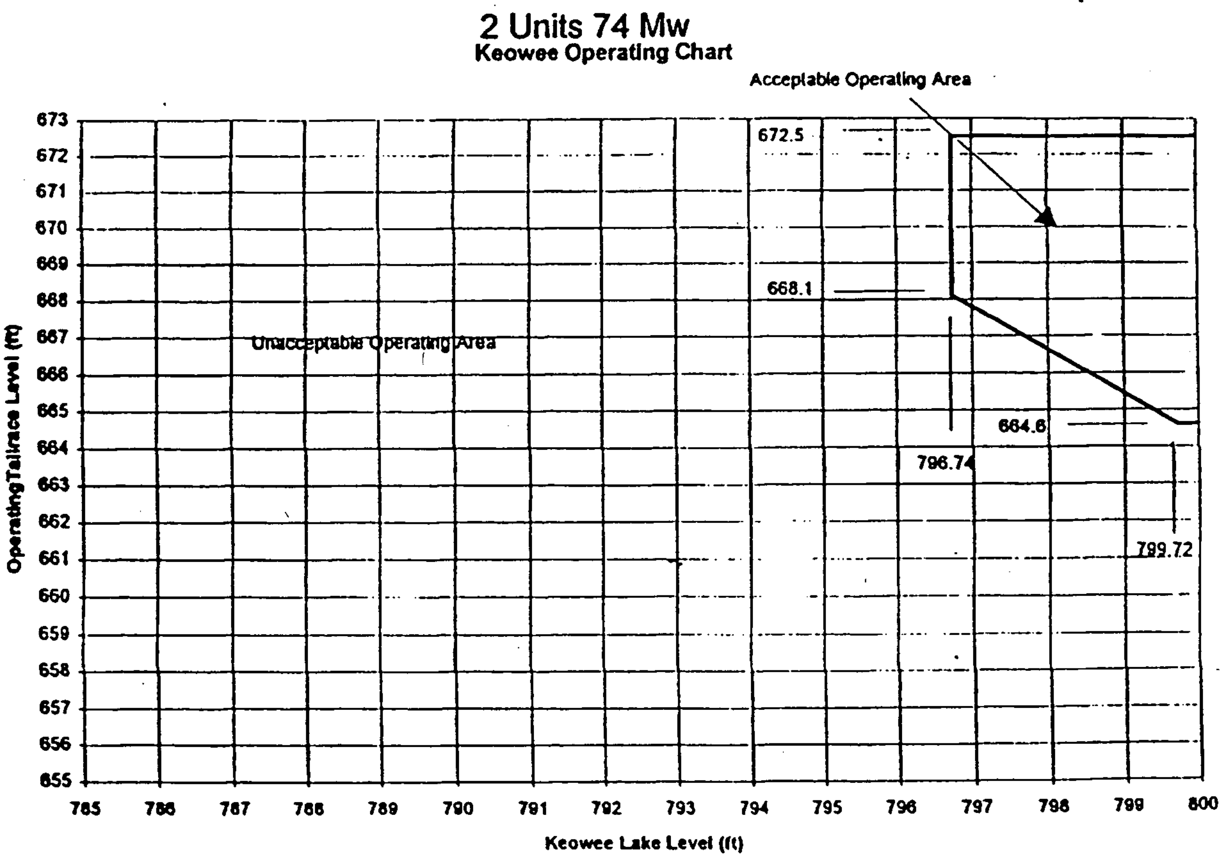
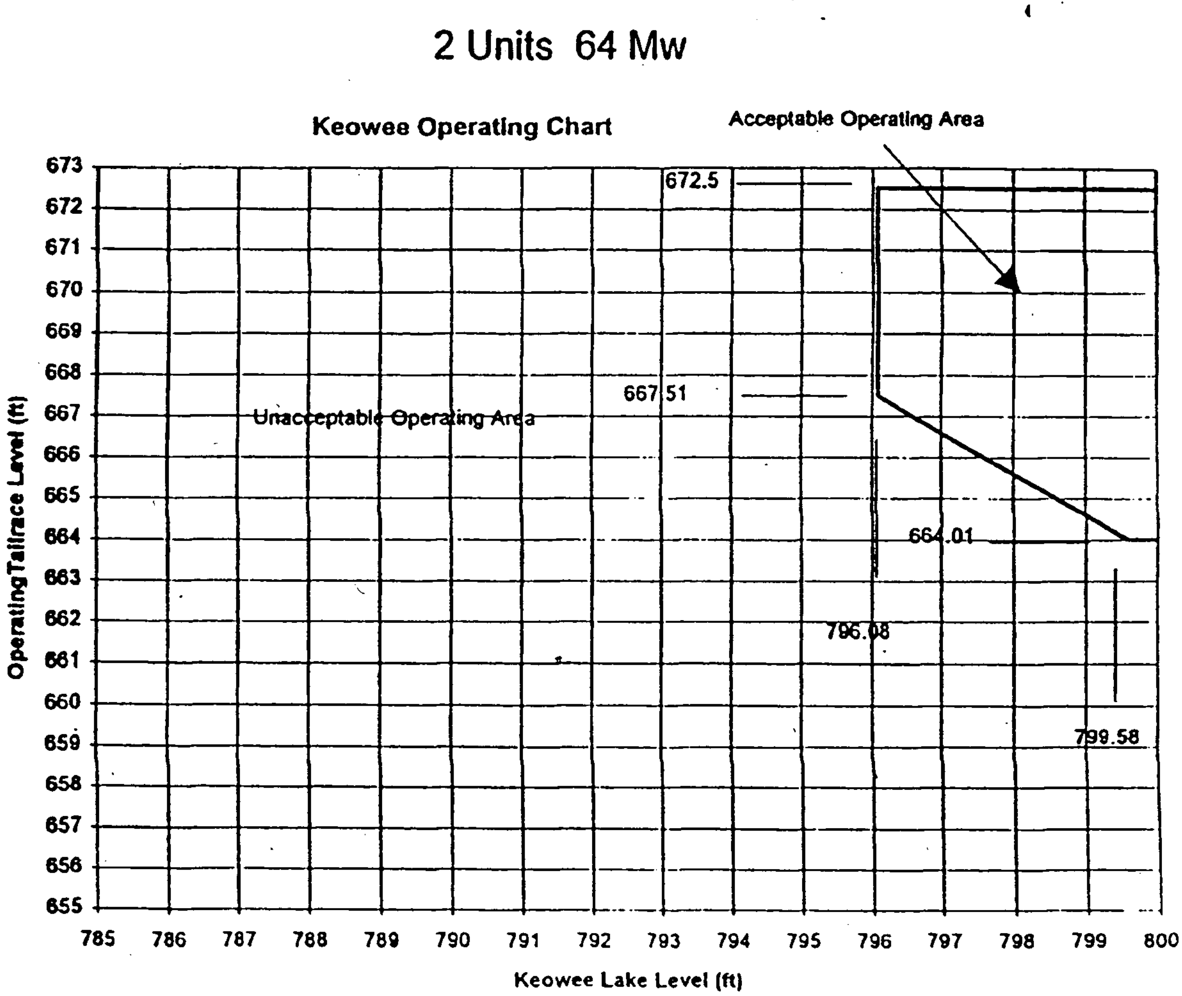


Figure 16.8.4-4
Keowee Operational Restrictions



BACKGROUND

Portions of this SLC are relocated from CTS 3.7.1 TS Note 3.

During periods of commercial power generation, the operability of the Keowee Hydro units shall be based on lake levels and the power level of the Keowee Hydro units. The Keowee Hydro operating restrictions for commercial power generation shall be contained in the ONS Selected Licensee Commitment manual.

This SLC is used to determine Keowee Hydro unit operability as an Oconee Emergency Power source when Keowee is generating to the commercial grid. It specifies the range of acceptable Keowee lake and tailrace elevations for various Keowee power generation levels. The acceptable region of the operating restrictions was determined by reference 1.

Figure 16.8.4-1 specifies the maximum operating limits of the Keowee Hydro units. This is applicable only for single unit operation of Keowee unit 1. This refers to occasions when Keowee unit 1 is operating and Keowee unit 2 is not operating. This figure allows for operation of Keowee Hydro unit 1 at a maximum of 84MW. Also, any operation of Keowee Hydro unit 1 below 84MW is allowed in accordance with this figure.

Figure 16.8.4-2 is applicable only for single unit operation of Keowee unit 2. This refers to occasions when Keowee unit 2 is operating and Keowee unit 1 is not operating. This figure allows for operation of Keowee Hydro unit 2 at a maximum of 84 MW. Also, any operation of Keowee Hydro unit 2 below 84 MW is allowed in accordance with this figure.

Figures 16.8.4-3 and 16.8.4-4 apply to simultaneous commercial generation with both Keowee units. In addition, the figures apply to single unit operation of Keowee unit 1 or 2. In Figure 16.8.4-3, commercial generation is allowed up to a maximum of 74MW. Figure 16.8.4-4 contains the operating restrictions for commercial generation up to a maximum of 64MW. The lake levels on the operating charts are operating lake levels. Therefore, verification that the operation of the Keowee Hydro units is within the acceptable region of the charts will have to be performed during operation of the Keowee Hydro units.

APPLICABLE SAFETY ANALYSIS

The Keowee Hydro units provide emergency power for Oconee Nuclear Station on the appropriate emergency power path. The operability of the Keowee Hydro units is required to ensure the operability and the capability of the Emergency Power System. Nuclear Station Modification (NSM) ON-52966 installed frequency protection and revised the runaway governor protection logic circuits which ensure the operability of the Keowee Hydro units during periods of commercial generation. This SLC will ensure that the Keowee Hydro units are operated within the acceptable limits.

APPLICABILITY

During periods of commercial power generation, the Keowee Hydro units are required to be within the acceptable regions of the operating restrictions when one or more Oconee Nuclear units are in MODES 1, 2, 3 or 4.

ACTIONS

The operability of the Keowee Hydro units during periods of commercial generation is ensured when the Keowee Hydro units operate within the acceptable region of Figures 16.8.4-1, 16.8.4-2, 16.8.4-3 and 16.8.4-4.

A.1

If the Keowee Hydro units are determined to be outside the limits of the acceptable region, action will be taken to restore commercial generation of the Keowee Hydro units to within the limits of the acceptable region. In addition, the applicable ITS Condition shall be entered since the Keowee Hydro Unit may not be able to perform its design function. Once the commercial operation of the Keowee Hydro unit(s) is restored to within the limits of the acceptable region, the ITS Condition shall be exited. It is not necessary to perform an operability test of Keowee Hydro units prior to exiting the Condition as long as no maintenance is performed on the units in order to return them to an acceptable operating region.

SURVEILLANCE REQUIREMENTS

SR 16.8.4.1

This surveillance will ensure that the operating conditions are within the limits of the acceptable region of the operating restrictions in Figure 16.8.4-1, 16.8.4-2, 16.8.4-3 and -16.8.4-4 during commercial generation by the Keowee Hydro units. Since the lake levels in Figures 16.8.4-1, 16.8.4-2, 16.8.4-3 and 16.8.4-4 are operating lake levels, verification that the operation of the Keowee Hydro units is within the acceptable regions will be performed during operation of the Keowee Hydro units.

REFERENCES:

1. Calculation KC-UNIT1-2-0106
2. 04/19/95 letter from J. W. Hampton to the NRC, "Response to NRC Questions on the Proposed Emergency Power Modification Action Plan."
3. 03/15/95 letter from J. W. Hampton to the NRC, "Proposed Emergency Power Modification Action Plan."
4. 08/15/95 letter from the NRC to J. W. Hampton, "Issuance of Amendments"
5. 03/20/97 Safety Evaluation Report from the NRC to add OPERABILITY requirements and surveillances to the Technical Specifications.

16.8 ELECTRIC POWER SYSTEM

16.8.5 125 Vdc Vital I&C System Ground Locating Policy

COMMITMENT Grounds on the 125 Vdc Vital Instrumentation and Control System will be pursued in accordance with this ground locating policy.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. A continuous ground alarm is received.	A.1 Determine the ground magnitude.	-----NOTE------ If the buses cannot be separated due to extenuating circumstances, then the ground magnitude will be determined within 8 hours once the buses are separated. -------------- 8 hours
B. One or more ground alarms are present. OR One ground alarm is inoperable.	B.1 Measure ground voltage and bus voltage.	------NOTE----- A 50% interval extension applies to the Completion Times. ------------- Once within 12 hours AND 12 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Ground resistance < 2.8V to ground (≤ 500 Ohms).	C.1 Initiate efforts to locate the ground.	24 hours from receipt of continuous ground alarm
	<u>AND</u>	
	C.2 Perform engineering evaluation of safety system vulnerability to the ground using the available data.	7 days from receipt of continuous ground alarm
	<u>AND</u>	
	C.3 Request PORC approval of the evaluation.	7 days from receipt of continuous ground alarm
D. Ground resistance ≥ 2.8V and < 6V (> 500 Ohms and ≤ 2,000 Ohms).	D.1 Initiate efforts to locate the ground.	48 hours from receipt of continuous ground alarm
	<u>AND</u>	
	D.2 If ground is not located, perform engineering evaluation of safety system vulnerability to the ground using the available data.	14 days from receipt of continuous ground alarm
	<u>AND</u>	
	D.3 Request PORC approval of the evaluation.	14 days from receipt of continuous ground alarm

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Ground resistance $\geq$ 6V and $\leq$ 18V (> 2,000 Ohms and $\leq$ 10,000 Ohms).	E.1 Initiate efforts to locate the ground.	128 hours from receipt of continuous ground alarm
	<u>AND</u>	
	E.2 If ground is not located, perform engineering evaluation of safety system vulnerability to the ground using the available data.	728 hours from receipt of continuous ground alarm
	<u>AND</u>	
	E.3 Request PORC approval of the evaluation.	728 hours from receipt of continuous ground alarm

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.8.5.1 NA	NA

BASES

BACKGROUND

The DC ground locating process was identified as a weakness in NRC inspection report 50-269,270,287/93-26. During December 1993, a pressure switch failed which resulted in a ground on the DC system and the inoperability of the 2A Motor Driven Emergency Feedwater Pump. This inoperability exceeded the allowed outage time in the CTS and resulted in NRC Violation 50-270/94-08-02. The response to this violation indicated that guidelines would be developed for locating a DC ground, evaluating the significance of the ground, and removing the ground. The guidelines have been developed and are contained in this SLC.

The primary concern with grounds is that the interaction of two or more grounds may cause the malfunction of equipment. The actions contained in this SLC are based on the recognition that lower resistance grounds have a greater probability to induce equipment malfunctions. In addition, the risk of ground-induced malfunctions is decreased as the ground resistance increases. This decrease in risk is associated with the lower probability of affecting individual relays and the reduced number of relays which are vulnerable.

APPLICABILITY

At all times, DC grounds on the 125Vdc Vital Instrumentation and Control System will be located in accordance with this ground locating policy.

ACTIONS

A.1

When a continuous valid ground alarm is received, Operations should evaluate the possible source of the ground. Information may be provided in the Alarm Response Procedure to aid in this evaluation. As part of the evaluation, a work request will be generated. SPOC/I&E Maintenance will take bus-to-ground Voltage measurements before the panelboards are separated. Following the completion of the voltage measurements, the panelboards will be separated. However, separation of the panelboards may not be allowed if Operations determines that extenuating circumstances exist. If the buses cannot be separated due to extenuating circumstances, then the ground magnitude will be determined within 8 hours once the buses are separated. The ground magnitude will be determined by SPOC/I&E Maintenance with the panelboards separated.

B.1

Whenever valid ground alarms are present or a ground alarm is inoperable, ground voltage measurements will be made every 12 hours. Any changes in the positive-to-ground or negative-to-ground voltages will be evaluated to ensure that there is no additional system degradation. In addition to the ground voltage measurements, the bus voltage measurements will be made to assure proper charger operation.

C.1

If the ground magnitude is determined to be ≤ 500 Ohms, then locating efforts will begin within 24 hours from receipt of the ground alarm. If determination of ground magnitude is delayed due to extenuating circumstances as described above, ground locating efforts will begin within 16 hours after the 8 hour period for determination of the ground magnitude. The ground will be located within 7 days after the receipt of a continuous ground alarm, or an evaluation of the safety system vulnerability using the available ground data will be performed. This 7 day action statement is based on 24 hours to initiate the locating efforts and 6 days to locate the ground or perform an evaluation. The Plant Operations Review Committee (PORC) will be contacted to approve the evaluation.

D.1

If the ground magnitude is determined to be > 500 Ohms and ≤ 2,000 Ohms, then fewer relays are vulnerable. Locating efforts will begin within 40 hours. This is based on a total time period of 48 hours from receipt of the ground alarm until locating efforts begin (40 hours plus the initial 8 hours). If determination of ground magnitude is delayed due to extenuating circumstances as described above, ground locating efforts will begin within 40 hours after the 8 hour period for determination of the ground magnitude. The ground will be located within 14 days after the buses are separated, or an evaluation of the safety system vulnerability using the available ground data will be performed. This 14 day action statement is based on 48 hours to initiate the locating efforts and 12 days to locate the ground or perform an evaluation. The Plant Operations Review Committee (PORC) will be contacted to approve the evaluation.

E.1

If the ground magnitude is determined to be > 2,000 Ohms and ≤ 10,000 Ohms, then locating efforts will begin within 128 hours. If determination of ground magnitude is delayed due to extenuating circumstances as described above, ground locating efforts will begin within 5 days after the 8 hour period for

determination of the ground magnitude. The ground will be located within 728 hours after receipt of a continuous ground alarm, or an evaluation of the safety system vulnerability using the available ground data will be performed. This 728 hour action statement is based on 5 days and 8 hours to initiate the locating efforts and 25 days to locate the ground or perform an evaluation. The Plant Operations Review Committee (PORC) will be contacted to approve the evaluation.

REFERENCES:

1. LER 270/94-01 dated March 10, 1994, "Technical Specification Limit Exceeded Due to Equipment Failure"
2. NRC Inspection report 50-269,270,287/93-26
3. NRC Inspection report 50-269,270,287/94-08
4. 5/11/94 letter from J. W. Hampton to NRC Document Control Desk, "Reply to Notice of Violation"
5. 6/23/94 letter from J. W. Hampton to NRC Document Control Desk, "Reply to Notice of Violation"
6. 2/9/95 memo from L. S. Underwood to C. A. Little, "DC Ground Locating Policy"

16.8 ELECTRIC POWER SYSTEM

16.8.6 Lee/Central Alternate Power System

COMMITMENT Two Lee Combustion Turbines (LCTs) shall be available for supplying power to the Oconee Standby Buses through a separated 100 kV power path within one hour of a loss of both On-Site Emergency Power Paths. Requirements for energizing the Oconee Standby buses are found in the ITS.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required LCTs not available for supplying power to the Standby Buses within one hour of demand.	A.1 --------NOTE-------- Lee/Central Power System is considered unavailable as input to Maintenance Rule Risk Assessment and Unavailability Monitoring. -------------------- Log unavailability in the Operations Log.	NA

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. CT-5 is not available within one hour of demand. __OR__ 100 kV power path from Lee is not available within one hour of demand. __OR__ Both SL breakers are not available within one hour of demand.	B.1 --------NOTE-------- Lee/Central Power System is considered unavailable as input to Maintenance Rule Risk Assessment and Unavailability Monitoring. -------------------- Log unavailability in the Operations Log.	NA
C. Power from a LCT is lost while supplying power to the Standby Buses. __OR__ Failure of a required LCT to start within one hour of demand.	C.1 --------NOTE---------- LCT is considered to have had a run failure as input into Maintenance Rule Failure Monitoring. ----------------------- Log unavailability in Operations Log.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.6.1	Verify with Lee Steam Station the availability status of required LCTs for supplying power to the standby buses.	24 hours

BASES

BACKGROUND

The three Lee Combustion Turbines 4C, 5C, and 6C; the separated 100 kv line through CT-5; and the SL Breakers serve as an alternate electrical power source for Oconee Nuclear Station when its on-site emergency power sources are unavailable. Only one Lee Combustion Turbine and one SL Breaker is required to supply hot shutdown loads for two Oconee units plus LOCA loads for one Oconee unit. The availability of two Lee Combustion Turbines allows for redundancy.

ACTIONS

When less than two Lee Combustion Turbines or its separated power path are available within one hour of the loss of both on-site emergency power paths, the Oconee units are more susceptible to a station blackout event (SBO). Adherence to Maintenance Rule Risk Assessment guidelines reduces the probability of a blackout event and increases the availability of SBO mitigation equipment. Unavailability of this equipment is logged in the Operations Log. Requirements for energizing the Oconee Standby busses are found in the ITS.

SURVEILLANCE REQUIREMENTS

The surveillance requires daily communication between Lee and Oconee, keeping Oconee personnel informed of the availability of the Lee Combustion Turbines for supplying power to the Oconee Standby Buses.

REFERENCES

1. Oconee Nuclear Station ITS 3.8
2. Work Process Manual Section 607, "Maintenance Rule Assessment of Equipment Removed From Service."
3. OSC-5771 "PRA Risk Significant SSC's for the Maintenance Rule."
4. OSS-0254.00-00-2011 100KV Alternate Power System Design Basis Document.

16.8 ELECTRIC POWER SYSTEMS

16.8.7 Auctioneering Diodes

COMMITMENT Perform specified SR.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.7.1	Verify peak inverse voltage capability of each I&C auctioneering diode is within limits.	184 days

BASES

The requirement(s) of this SLC section were relocated from CTS SR 3.7.8.2 during the conversion to ITS.

Each panelboard receives its DC power through an auctioneering network of two isolating diode assemblies. One assembly is supplied from the unit's 125 volt distribution system, and the other assembly is supplied from another unit's (the backup unit) 125VDC Vital Distribution System. The diode assemblies permit the two distribution systems to supply current to the Vital I&C DC Panelboard connected to the output of the diode assemblies, and block the flow of current from one DC distribution system to the other. Measuring peak inverse voltage capability of each auctioneering diode ensures the diodes are capable of isolating a fault on one source from the other source. The 184 day frequency is based on engineering judgement and operating experience.

REFERENCES

N/A

16.8 ELECTRIC POWER SYSTEMS

16.8.8 External Grid Trouble Protection System

COMMITMENT Perform specified SR.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.8.1	Verify the External Grid Trouble Protection System logic provides an isolated power path between Keowee and Oconee.	92 days

BASES

The requirement(s) of this SLC section were relocated from CTS 4.6.5 during the rewrite of CTS 3.7 (Amendment Nos. 232, 232, 231).

REFERENCES

N/A

16.9 AUXILIARY SYSTEMS

16.9.1 Fire Suppression Water Supply Systems

COMMITMENT The Fire Suppression Water Supply Systems shall be OPERABLE as follows:

<u>Oconee</u>

a. High Pressure Service Water (HPSW) pumps A and B with automatic initiation logic, and associated piping and valves supplying water to the sprinkler system and fire hose stations.

b. The HPSW pumps shall be aligned to the high pressure fire header.

<u>Keowee</u>

c. The Fire Protection Pump, automatic initiation logic. the associated piping and valves supplying water to the Main Transformer water spray system and hose stations listed in SLC 16.9.4 with the exception of the Mechanical Equipment Gallery stations

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Equipment inoperable in the Oconee Fire Suppression Water Supply System.	A.1 Restore inoperable equipment to OPERABLE status.	7 days
	<u>OR</u>	
	A.2 Develop guidance outlining plans and procedures to be used to compensate for the loss of redundancy in this system.	7 days
B. Equipment inoperable in the Keowee Fire Suppression Water Supply System.	B.1 Restore inoperable equipment to OPERABLE status.	7 days
	<u>OR</u>	
	B.2 Develop guidance outlining plans and procedures to be used to compensate for the loss of redundancy in this system.	7 days
C. No Oconee Fire Suppression Water Supply System OPERABLE.	C.1 Establish backup Oconee Fire Suppression Water Supply System.	24 hours
D. Required Action and associated Completion Time not met for Condition C.	D.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2 Be in MODE 5.	60 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. No Keowee Fire Suppression Water Supply System OPERABLE.	E.1 Establish backup Keowee Fire Suppression Water Supply System.	24 hours
	<u>OR</u>	
	E.2 Develop plan to restore Keowee primary Fire Suppression Water Supply System.	48 hours
F. Required Action and associated Completion Time not met for Condition E.	F.1 Energize both Standby Buses from Lee Combustion Turbine on the dedicated line.	1 hour
	<u>AND</u>	
	F.2 Restore Fire Suppression Water Supply System or alternate capability.	14 days
G. Required Action and associated Completion Time not met for Condition F.	G.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	G.2 Be in MODE 5.	60 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.1.1	Functionally test the Keowee Fire Protection Pump.	7 days
SR 16.9.1.2	Functionally test the Oconee HPSW pumps, power supplies and associated automatic valve.	31 days
SR 16.9.1.3	Verify proper alignment of valves for Keowee and Oconee.	31 days
SR 16.9.1.4	Verify flow for the Oconee HPSW pumps.	12 months
SR 16.9.1.5	Perform a performance test of the Keowee Fire Protection Pump.	12 months
SR 16.9.1.6	Flow test the Keowee Fire Water Suppression System by actuation of the Main Transformer water spray system.	12 months
SR 16.9.1.7	Perform Oconee Fire Suppression Water Supply System flow test in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14 Edition, NFPA.	36 months

BASES

Portions of Surveillance SR 16.9.1.2 involving the HPSW pumps and power supplies were relocated from CTS Table 4.1-2, Item 8 during the conversion to the ITS.

The OPERABILITY of the Fire Suppression System ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System Consists of the Water Supply System, spray and/or sprinklers, Keowee CO_2 and fire hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program. In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

The Testing Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. In the event the Fire Suppression Water Supply System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the Provisions of Oconee Facility Operating License conditions.

REFERENCES:

1) Oconee UFSAR, Chapter 9.5.1
2) Oconee Fire Protection SER dated August 11, 1978
3) Oconee Fire Protection Review, as revised.
4) Oconee Plant Design Basis Specification for Fire Protection as revised.

16.9 AUXILIARY SYSTEMS

16.9.2 Sprinkler And Spray Systems

COMMITMENT Sprinkler and Spray Systems in safety related areas listed in Table 16.9.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Sprinkler or Spray Systems inoperable. AND Affected Area(s) has no OPERABLE fire detection.	A.1 Establish continuous fire watch with backup fire suppression equipment in the area.	1 hour
B. One or more required Sprinkler or Spray Systems inoperable. AND Affected Area(s) has OPERABLE fire detection.	B.1 Establish hourly fire watch with backup fire suppression equipment in the area.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.2.1	-----------------NOTE-------------------- Not required to be performed for systems in the cable spreading room, equipment rooms and cable shafts. --- Functionally test each required Sprinkler or Spray System.	 12 months
SR 16.9.2.2	Inspect each required Sprinkler Systems spray headers and nozzles.	12 months
SR 16.9.2.3	Verify by visual inspection each nozzle's spray area to ensure spray pattern is not obstructed.	18 months

Table 16.9.2-1
Sprinkler and Spray Systems

a. Oconee Nuclear Station

i.	Turbine Driven Emergency FDW Pump	Units 1, 2, and 3
ii.	Transformers[1]	CT-1, CT-2, CT-3, CT-4, CT-5
iii.	Cable Room	Units 1, 2, and 3
iv.	Equipment Room	Units 1, 2, and 3
v.	Cable Shaft (3rd Level)	Units 1, 2, and 3
vi.	Cable Shaft (4th & 5th Level)	Units 1, 2, and 3

b. Keowee Hydro Station

i. Main Lube Oil Storage Room

ii. Main Transformer

1. The transformers do not have fire detection devices. They have Activation devices that actuate the deluge valve of the fire suppression systems only.

BASES

The OPERABILITY of the NRC committed Fire Suppression System ensures that adequate fire suppression capability is available to confine and extinguish fires occurring at the Oconee or Keowee facilities. The regulatory requirement is to have NRC committed Sprinkler and Spray Systems OPERABLE only when the equipment it is protecting is required OPERABLE for plant safety. However, to protect the equipment for property conservation and minimize equipment loss due to fire; the Oconee and Keowee NRC committed Sprinkler and Spray Systems will be rquired to be OPERABLE at all times.

The Oconee CT-1, 2, 3, and 5 transformers do not have fire detection devices. They have fire actuation devices that actuate the deluge valve of the fire suppression systems. These actuation devices do not directly annuciate to the Control Rooms. When the deluge valve trips, the flow pressure switch is the sensor that activiates the Control Room alarms. With HPSW deactivated for maintenance or testing, there is no form of annucation of a fire in the Control Room.

During periods of time when the Sprinkler or Spray system is not operable and detection instrumentation is operable, a hourly fire watch patrol will be required to inspect the affected area frequently as a precaution. If the sprinkler or spray system in the area is not operable and no detection instrumentation is operable, a continuous fire watch is required to be maintained in the vicinity of the affected sprinkler or spray system until the system is restored to operable status.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

The test requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License conditions.

REFERENCES

1. Oconee UFSAR, Chapter 9.5-1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, as revised.
4. Oconee Plant Design Basis Specification for Fire Protection, as revised.

16.9 AUXILIARY SYSTEMS

16.9.3 Keowee CO_2 Systems

COMMITMENT The automatic CO_2 system provided for the generators at Keowee Hydro Station shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Keowee CO_2 System inoperable.	A.1 Establish continuous fire watch with backup fire suppression equipment in the area.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.3.1	Verify each valve in the flow path is in its correct position.	31 days
SR 16.9.3.2	Verify CO_2 storage tank weight is $\geq$ 90% full charge weight.	184 days
SR 16.9.3.3	Verify system actuates manually and automatically upon receipt of a simulated action signal.	18 month
SR 16.9.3.4	Perform flow test through headers and nozzles to assure no blockage.	18 months

BASES

The OPERABILITY of the NRC committed Fire Suppression system ensures that adequate fire suppression capability is available to protect safety-related equipment by confining and extinguishing fires occurring in the Keowee eletric generators. The Fire Suppression System consists of the water system, spray and/or sprinklers, Keowee CO_2 system and fire hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. The Testing Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License Conditions.

REFERENCES:

1. Oconee UFSAR, Chapter 9.5-1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, as revised.
4. Oconee Plant Design Basis Specification for Fire Protection, as revised.

16.9 AUXILIARY SYSTEMS

16.9.4 Fire Hose Stations

COMMITMENT The Fire Hose Stations listed in Table 16.9.4-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Fire Hose Station outside reactor building inoperable.	A.1 Provide additional equivalent capacity fire hose of length to reach unprotected area at OPERABLE hose station.	1 hour
B. Required Fire Hose Station inside reactor building inoperable (water not available to isolation valves LPSW-563 and LPSW-564).	B.1 Ensure availability of 4 portable fire extinguishers outside containment in the personnel hatch area of the auxiliary building for fire brigade use upon entering reactor building.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.4.1	Perform visual inspection, including inspection of coupling gaskets, of the fire hose stations located outside the reactor building and inside reactor building that are accessible during power operation.	31 days
SR 16.9.4.2	Perform visual inspection, including inspection of coupling gaskets, of reactor building fire hose stations that are inaccessible during power operation.	18 months
SR 16.9.4.3	Partially stroke test Fire Hose Station Valves.	36 months
SR 16.9.4.4	Subject each fire hose to hydrostatic test at pressure $\geq$ 50 psig greater than the maximum pressure at the station.	36 months
SR 16.9.4.5	Perform maintenance inspection including removal and reracking the hoses and inspection of coupling gaskets.	36 months

Table 16.9.4-1
Fire Hose Stations

a. Oconee Nuclear Station

Location No.	Valve No.	Area or Component Protected
3-D-28	2HPSW-194	1&2 Blockhouse, 1 & 2 3rd Floor Switchgear
AX-35	1HPSW-436	#1 Cable Spread Room
AX-32	2HPSW-436	#2 Cable Spread Room
AX-33	2HPSW-437	1 & 2 Cable Spread Room
AX-30	3HPSW-436	#3 Cable Spread Room
AX-31	3HPSW-437	#3 Cable Spread Room
5-M-31	2HPSW-304	1 & 2 Control Room, 1 & 2 Emergency Shutdown Panels
TOH-3	3HPSW-338	#3 Control Room, #3 Emergency Shutdown Panels
1-J-28	2HPSW-242	#1 First Floor MCCs HPSW Pumps, 1 & 2 LPSW Pumps
1-J-43	3HPSW-344	#3 1st Floor Motor Control Canters
1-B-19	1HPSW-283	#1 EFWP
1-D-39	2HPSW-236	#2 EFWP
1-D-53	3HPSW-336	#3 EFWP
AX-13	1HPSW-448	1 & 2 HP1 Pumps, 1 & 2 LPI Pumps
AX-14	3HPSW-449	3 HP1 Pumps, 3 LPI Pumps
1-J-47	3HPSW-348	3 LPSW Pumps
AX-36	1HPSW-445	#1 West Penetration Room
AX-45	1HPSW-444	#1 East Penetration Room
AX-42	2HPSW-444	#2 East Penetration Room
AX-43	2HPSW-445	#2 West Penetration Room
AX-29	3HPSW-444	#3 East Penetration Room
AX-44	3HPSW-445	#3 West Penetration Room
AX-21	HPSW-457	1 & 2 Equipment Room
AX-19	3HPSW-458	3 Equipment Room
3-M-24	HPSW-176	1 Equipment Room
3-M-29	2HPSW-245	2 Equipment Room
3-M-43	3HPSW-339	3 Equipment Room
3-J-28	2HPSW-241	1 & 2 3rd Floor Switchgear
3-M-43	3HPSW-339	3 3rd Floor Switchgear, 600V Load Center
AX-22	1HPSW-440	1 Battery Room
AX-20	2HPSW-440	2 Battery Room
AX-18	3HPSW-440	3 Battery Room
1RBH1	1LPSW-471	Ground Floor Level - East Side
2RBH1	2LPSW-471	Basement Floor Level - East Side
3RBH1	3LPSW-471	Basement - East side
1RBH2	1LPSW-473	Intermediate Floor Level - East Side
2RBH2	2LPSW-473	Intermediate Floor Level - East Side
3RBH2	3LPSW-473	Intermediate Floor Level - East Side
1RBH3	1LPSW-475	Top of Shielding Floor Level - East Side
2RBH3	2LPSW-475	Top of Shielding Floor Level - East Side
3RBH3	3LPSW-475	Top of Shielding Floor Level - East Side
1RBH4	1LPSW-465	Top of Shielding Floor Level - West Side
2RBH4	2LPSW-465	Top of Shielding Floor Level - West Side
3RBH4	3LPSW-465	Top of Shielding Floor Level - West Side
1RBH5	1LPSW-467	Intermediate Floor Level - West Side

Table 16.9.4-1
Fire Hose Stations

Location No.	Valve No.	Area or Component Protected
2RBH5	2LPSW-467	Intermediate Floor Level - West Side
3RBH5	3LPSW-467	Intermediate Floor Level - West Side
1RBH6	1LPSW-469	Ground Floor Level - West Side
2RBH6	2LPSW-469	Basement Floor Level - West Side
3RBH6	3LPSW-469	Basement - West Side
VBH-1	HPSW-916	Essential Siphon Vacuum Building
VBH-2	HPSW-917	Essential Siphon Vacuum Building
Basement	-	EL. 777' 6"
Ground	-	EL. 797' 6"
Intermediate	-	EL. 825' 0"
Top of Shielding	-	EL. 861' 0"

b. Keowee Hydro Station

Location No.	Valve No.	Area or Component Protected
Operating Deck (NW)	KH-1	Operating Floor
Operating Deck (NE)	KH-2	Operating Floor
Operating Deck (SW)	KH-4	Operating Floor
Operating Deck (SE)	KH-3	Operating Floor
Control Room	KH-6	Control Room
Mech. Equip. Gallery	KH-5	Mech. Equip. Gallery

BASES

The OPERABILITY of the NRC committed Fire Suppression System ensures that adequate fire suppression capability is available to confine and extinguish fires occurring at the Oconee or Keowee facilities. The regulatory requirement is to have NRC committed Fire Hose Stations OPERABLE only when the equipment it is protecting is required OPERABLE for plant safety. However, to protect the equipment for property conservation and minimize equipment loss due to fire; the Oconee and Keowee NRC committed Fire Hose Stations will be rquired to be OPERABLE at all times.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available for the affected areas until the inoperable equipment is restored to service.

The testing requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression System are met.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License Conditions.

REFERENCES:

1. Oconee UFSAR, Chapter 9.5-1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, as revised.
4. Oconee Plant Design Basis Specification for Fire Protection, as required.

16.9 AUXILIARY SYSTEMS

16.9.5 Fire Barriers

COMMITMENT All Fire Barriers (including mechanical and electrical penetrations, fire doors, fire dampers, walls, ceilings and floors) boundaries, as shown on the 0-310-K and 0-310-L series drawings shall be operable.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Fire Barrier inoperable. AND There is OPERABLE fire detection instrumentation within 15 feet on both sides of the fire barrier boundary inoperability location.	A.1 Determine OPERABILITY status of fire detection instrumentation for the affected area(s). AND A.2 Establish an hourly fire watch patrol on at least one side of the fire boundary.	1 hour 1 hour
B. Required Fire Barrier inoperable. AND Affected Area(s) has OPERABLE fire detection instrumentation within 15 feet on only one side of the fire barrier boundary inoperability location.	B.1 Determine OPERABILITY status of fire detection instrumentation for the affected area(s). AND B.2 Establish hourly fire watch patrol in the area that does not have OPERABLE fire detection instrumentation.	1 hour 1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Fire Barrier inoperable. AND Affected Area(s) has no OPERABLE fire detection instrumentation within 15 feet.	C.1 Establish continuous fire watch on one side of affected penetration fire barrier.	1 hour

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 16.9.5.1	Visually inspect exposed surfaces of each fire rated barrier.	18 months
SR 16.9.5.2	Visually inspect at least 10% of all fire dampers. If apparent changes in appearance or abnormal degradation is found, a visual inspection of an additional 10% of the dampers shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each fire damper will be inspected every 15 years.	18 months
SR 16.9.5.3	Visually inspect at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.	18 months

BASES

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections and sampling.

The OPERABILITY of a NRC committed fire barrier ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. The regulatory requirement is to have NRC committed Fire Barriers OPERABLE only when the equipment it is protecting is required OPERABLE for plant safety. However, to also protect the equipment for property conservation and minimize equipment loss due to fire; the Oconee and Keowee NRC committed Fire Barriers will be required to be OPERABLE at all times.

During periods of time when a barrier is not functional, a fire watch patrol will be required to inspect the affected area frequently as a precaution in addition to the fire detection instrumentation in the area. If fire detection instrumentation in the area is not operable, a continuous fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status. Fire detection is not specifically designed at ONS to provide early detection of fire near the committed fire boundaries as denoted on the drawing series 0-310K and 0-310L. Fire detection instrumentation design locations were typically based on protecting specific equipment and areas important to safety or where major fire hazards were located and not to provide full detection coverage for all areas of the plant. In actuality fire detection instrumentation's ability to quickly respond to the incipient stages of a fire are based on distance from the hazard, type of hazard, obstruction, and air flows in the area. The application of the specific fire detection instrumentation used at ONS provide a adequate response time for a floor distance of approximately 15 feet in radius from the detector location.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License Conditions.

REFERENCES:

1. Oconee UFSAR, Chapter 9.5-1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, as revised.
4. Oconee Plant Design Basis Specification for Fire Protection, as revised.

16.9 AUXILIARY SYSTEMS

16.9.6 Fire Detection Instrumentation

COMMITMENT The provided Fire Detection Instrumentation for each equipment/location shall be OPERABLE as listed in Table 16.9.6-1.

--------------------------NOTE--------------------------
Fire Detection Instrumentation located within containment is not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

APPLICABILITY: At all times.

ACTIONS

---NOTE---

OPERABILITY of fire detection instrumentation for adequate equipment/location coverage may also be determined by the Site Fire Protection Engineer or designee.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. > 50% of required detectors for one or more Oconee equipment/location inoperable. OR 2 required adjacent detectors for one or more Oconee equipment/location inoperable.	A.1 ---NOTE--- An hourly firewatch is not required for inaccessible equipment/locations such as the Reactor Building at power operation. Periodic inspections using a TV camera (if available) are permitted as described in Site Directives, or, the inaccessible equipment condition may be monitored by remote indications which would provide early warning of a fire. --- Establish hourly fire watch patrol (or as permitted by Site Directives) to inspect the accessible area with the inoperable instrumentation.	 1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. > 50% of required detectors for one or more Keowee equipment/location inoperable. <u>OR</u> 2 required adjacent detectors for one or more Keowee equipment/location inoperable.	B.1 Establish hourly fire watch patrol to inspect the accessible area with the inoperable instrumentation.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.6.1	Perform CHANNEL FUNCTIONAL TEST of Oconee Fire Detection Instruments using Fire Detection Instrumentation Control Board Panel Test Switch.	31 days
SR 16.9.6.2	Visually inspect Oconee Fire Detection Instruments accessible during power operation.	184 days
SR 16.9.6.3	Visually inspect Keowee Fire Detection Instruments.	184 days
SR 16.9.6.4	Test each Oconee fire detector for sensitivity.	12 months
SR 16.9.6.5	Perform CHANNEL FUNCTIONAL TEST of Keowee Fire Detection Instruments.	12 months
SR 16.9.6.6	------------------NOTE-------------------- Not required to be performed for Keowee Generator Detectors. -- Test each Keowee fire detector for sensitivity.	12 months
SR 16.9.6.7	Visually inspect Oconee Fire Detection Instruments not accessible during power operation.	18 months

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

OCONEE NUCLEAR STATION

Units 1, 2, and 3 Reactor Buildings

Equipment	Detectors Provided
Reactor Building Penetrations	8 (each unit)
Reactor Building Cooling Units	6 (each unit)
Reactor Coolant Pumps	8 (each unit)

Units 1, 2, and 3 Auxiliary Building
EL. 822' +0

Room No.	Equipment	Detectors Provided
71-Q	Unit 1 Cable Shaft	2
510	Unit 1 and 2 Control Room	10
75-Q	Unit 2 Cable Shaft	2
552	Unit 3 Control Room	8
90-Q	Unit 3 Cable Shaft	2

EL. 809' + 3"

Room No.	Equipment	Detectors Provided
400	Unit 1 Control Battery Room	5
402	Unit 1 East Penetration Room	12
403	Unit 1 Cable Room and Cable Shaft	19
404	Unit 2 Cable Room and Cable Shaft	18
407	Unit 2 East Penetration Room	20
408	Unit 2 Control Battery Room	5
409	Unit 1 West Penetration Room	5
410	Unit 2 West Penetration Room	5
450	Unit 3 Cable Room	28
452	Unit 3 East Penetration Room	10
455	Unit 3 Ventilation Equipment	2
456	Unit 3 West Penetration Room	5
458	Unit 3 Control Battery Room	2

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

EL. 796' +6"

Room No.	Equipment	Detectors Provided
300	Unit 1 Work Area	9
310	Unit 1 Equipment Room and Cable Shaft	13
311	Unit 2 Equipment Room and Cable Shaft	15
313	Janitor's Closet (Unit 1)	1
314	Clean Protective Clothing Storage (Unit 1)	1
322	Protective Clothes Storage (Unit 2)	1
329	Hot Lab	1
330	Cold Lab	1
331	Counting Room (Unit 2)	1
333	Health Physics (Unit 2)	1
334	Office (Unit 2)	1
335	Environmental Lab (Unit 2)	1
337	Laundry Sorting (Unit 2)	1
338	Laundry Storage (Unit 2)	1
339	Laundry (Unit 2)	2
347	Work Area (Unit 2)	8
354	Unit 3 Equipment Room and Cable Shaft	21
357	Janitor's Storage (Unit 3)	1
364	Towel Storage (Unit 3)	1
365	Janitor's Storage (Unit 3)	1
366	Protective Clothing (Unit 3)	1
369	HP Office (Unit 3)	1
369A	Supv. Technicians Office	1
369B	Secondary Chemistry Lab	1
369C	I.C. Computer	1
376	Unit 3 Work Area	10

EL. 771' + 0

Room No.	Equipment	Detectors Provided
119	Unit 1 and 2 LPI Hatch Area	3
159	Unit 3 LPI Hatch Area	2

EL. 838"+0

Room No.	Equipment	Detectors Provided
611	Protective Clothing Storage (Unit 2)	1
658	Protective Clothing Storage (Unit 3)	1

EL. 783' + 9"

Room	Equipment	Detectors Provided
204	Storage (Unit 1)	1
207	Chemical Handling and Storage (Unit 1)	1
220	Hot Instrument Shop (Unit 2)	1
224	Storage (Unit 2)	1
264	Storage (Unit 3)	1

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

EL. 758' +0

Room No.	Equipment	Detectors Provided
54	Unit 1 High Pressure Injection Pumps	1
56	Unit 1 and 2 High Pressure Injection Pumps	1
58	Unit 2 High Pressure Injection Pumps	1
61	Unit 1 Low Pressure Injection Pumps	2
62	Unit 1 and 2 Low Pressure Injection Pumps	2
63	Unit 2 Low Pressure Injection Pumps	2
76	Unit 3 High Pressure Injection Pumps	1
77	Unit 3 High Pressure Injection Pumps	1
81	Unit 3 Low Pressure Injection Pumps	2
82	Unit 3 Low Pressure Injection Pumps	2

Units 1.2. and 3 Turbine Buildings
EL. 775'+0

Equipment	Detectors Provided
MCC 1XC. 1XD. 1XE. 1XF; Unit 1 FDW Turbines; Unit 1 Emergency Feedwater Turbine; Unit 1 H2 Panel; Unit 1 EHC Unit	10
MCC 2XB. 2XC. 2XD. 2XE. 2XF; Unit 2 FDW Turbine; Unit 2 Emergency Feedwater Turbine; Unit 2 H2 Panel; Unit 2 EHC Unit	11
MCC 3XC. 3XD. 3XE. 3XF; Unit 3 FDW Turbines; Unit 3 Emergency Feedwater Turbine; Unit 3 H2 Panel; Unit 3 EHC Unit	10

EL. 796' + 6"

Equipment	Detectors Provided
Switchgear 1TA. 1TB. 2TA. 2TB; Load Centers 1X1. 1X2. 1X3. 1X4. 1X5. 1X6. 2X1. 2X2. 2X3. 2X4. 2X5. 2X6	8
Switchgear B1T. B2T; Transformer CT4	5
Switchgear 3B1T. 3B2T	3
MCC 1XA	1
ITTC5 and ITTC6	1
Unit 1 Main Turbine Oil Tank	1
Unit 2 Main Turbine Oil Tank	1
Unit 3 Main Turbine Oil Tank	2
DC Distribution Center IDA; Switchgear 1TC. 1TD. 1TE	7
MCC 1XGA	1
DC Distribution Center 2DA; Switchgear 2TC. 2TD. 2TE	7
MCC 3XGA	1
MCC 2XGB	1
Load Center 3X1. 3X2. 3X3. 3X4; MCC 3XGA; Switchgear 3TC. 3TD. 3TE	5
MCC 3XGB	1

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

EL. 822' + 0

Equipment	Detectors Provided
Bearing Oil Lift Pumps for All Units	4 ea unit
High Pressure Unit for All Units	2 ea unit

KEOWEE HYDRO STATION

Equipment	Detectors Provided
Control Room	4
Battery Room	4
Mechanical Equipment Gallery	3
Main Lube Oil Storage Room	1
Generators 1 and 2	6 ea
Operating Floor	6

ESSENTIAL SIPHON VACUUM BUILDING	6

BASES

OPERABILITY of the NRC committed Fire Detection Instrumentation ensures that adequate warning capability is available for the prompt detection of fires in areas containing safety related and important to safety equipment at Oconee and Keowee Facilities. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program. The regulatory requirement is to have NRC committed Fire Detection Instrumentation OPERABLE only when the equipment it is protecting is required OPERABLE for plant safety. However, to also protect the equipment for property conservation and minimize equipment loss due to fire; the Oconee and Keowee NRC committed Fire Detection Instrumentation will be required to be OPERABLE at all times.

In the event that a portion of the Fire Detection Instrumentation is inoperable, the establishment of compensatory actions in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to operability.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License Conditions.

REFERENCES:

1. Oconee UFSAR, Chapter 9.5-1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, as revised.
4. Oconee Plant Design Basis Specification for Fire Protection, as revised.
5. Oconee Plant Design Basis Specification for Fire Detection, as revised.

16.9 AUXILIARY SYSTEMS

16.9.7 Keowee Lake Level

COMMITMENT a. Keowee lake level shall be > 794.15 ft to ensure that the requirements of ITS 3.7.7 (LPSW System) are met for all three Units.

b. Maintain lake level ≥ 784.15 ft to assure that the Keowee Oil Storage Room Water Spray System shall be OPERABLE.

c. Maintain lake level ≥ 781.15 ft to assure that adequate water supply shall be available for 7 days of Keowee emergency operation.

d. Maintain lake level ≥ 780.60 ft to assure that the Keowee Step-up Transformer Mulsifyre System shall be OPERABLE.

----------------------------NOTES------------------------------

1. The requirements of Commitment a do not apply in MODE 5.

2. Commitment d does not apply in MODE 5 when the Keowee step up transformer is not required to be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4,
MODE 5 when a Keowee Hydro Unit (KHU) is required to be OPERABLE

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Keowee Lake Level < 794.15 ft.	A.1	Enter applicable Condition for one required LPSW pump inoperable in accordance with ITS 3.7.7.	Immediately
B. Keowee Lake Level < 784.15 ft.	B.1	Declare the Keowee Oil Storage Room Water Spray System inoperable.	Immediately
C. Keowee Lake Level < 781.15 ft.	C.1	Cease commercial power generation using KHUs.	Immediately
	<u>AND</u>		
	C.2	Notify the Plant Operations Review Committee (PORC) per NSD-308 and Request plant operation (and reportability) guidance.	Immediately
D. Keowee Lake Level < 780.6 ft.	D.1	Declare Keowee Step-up transformer Mulsifyre inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 16.9.7.1	Verify Keowee lake level is within limits.	12 hours

BASES:

An instrument error of 1.15 ft has been added to the absolute lake level to obtain the indicated lake levels identified in this SLC. The indicated lake levels in this SLC are based on the use of a computer point to verify level. Absolute lake level can be determined at the Keowee Hydro intake structure.

LPSW System Commitments:

With lake level below 794.15 ft, calculations show that the LPSW pumps could experience inadequate NPSH with assisted siphon flow if a single failure causes only the minimum number of LPSW pumps (two for the shared Unit 1 and 2 LPSW System) to be available during a design basis event. Therefore, the LPSW system must be considered unable to withstand a single failure for lake level below 794.15 ft and a 72 hour ACTION must be entered per ITS 3.7.7 by declaring one required LPSW pump inoperable.

Keowee Oil Storage Room Commitment:

Should lake level fall below 784.15 ft, the Keowee Oil Storage Room water spray system may not provide the required flow rates because the system is dependent on lake level for driving head. For this reason, the spray system should be declared inoperable.

Keowee Hydro Station Commitment:

With lake level below 781.15 ft, the water supply (for Keowee Hydro Station to provide emergency power to the overhead path at 46.5 MVA and the underground path at 22.35 MVA) could be inadequate for 7 days of continuous operation at these levels. Neither Keowee Hydro or Oconee Nuclear Station should be considered inoperable at this lake level. Keowee Hydro should not generate to the grid at lake levels below 781.15 ft in order to ensure ample water capacity for emergency power operation.

Keowee Main Start-up Transformer Commitment:

Should lake level fall below 780.60 ft, the Keowee main Step up Transformer Mulsifyre system may not provide the required flow rates because the system is dependent upon lake level for driving head. For this reason, the Mulsifyre should be declared inoperable.

REFERENCES:

1. PIR 0 092 0535, Potential Insufficient NPSH for LPSW pumps
2. LER 269/93 04, Rev. 0 and Rev. 1
3. OSS-0254.00-00-1039, Rev. 10, Design Basis Specification for the LPSW System
4. Calculation OSC 2895, Rev. 4, Hydraulic Calculations for Keowee Deluge Systems
5. Calculation OSC 5325, Rev. 0, Keowee Lake Level Uncertainty Calculation
6. Calculation OSC 5022, Rev. 1, USQ Evaluation for Operability Evaluation of PIR 0-092-0535
7. Calculation OSC 2280, Rev. 10, LPSW NPSH and Minimum Required Lake Level
8. Calculation OSC-3528, Rev. 3, Keowee Lake Level Minimum Administrative Limits
9. ITS 3.7.8, Emergency Condenser Circulating Water, Ammendment Nos. 300/300/300.

16.9 AUXILIARY SYSTEMS

16.9.8 [DELETED]

16.9 AUXILIARY SYSTEMS

16.9.8a HPSW Requirements to Support Loss of LPSW

COMMITMENT HPSW should be available to support loss of LPSW as follows:

a. HPSW should be available to provide backup cooling water to the Turbine Driven Emergency Feedwater Pump.

b. HPSW should be available to provide the backup cooling water to HPI Pump motor coolers.

APPLICABILITY: Any time the Turbine Driven Emergency Feedwater Pump or an HPI pump is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. HPSW not available.	A.1 Operations perform Risk Assessment considering equipment out of service.	NA
	AND	
	A.2 Log unavailability duration in the Operations Log for Maintenance Rule performance monitoring.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.9.8a.1 NA.	NA

BASES

Surveillance per SLC 16.9.1 and the Appendix B testing program is adequate to demonstrate the availability of the equipment and systems discussed here.

For loss of all AC power (Station Blackout, SBO), HPSW will provide cooling water to the Turbine Driven Emergency Feedwater Pump. However, the Standby Shutdown Facility (SSF) remains the licensing and design basis commitment for decay heat removal during a Station Blackout. (Ref. 2, Section 20.1.2; Ref. 4, 7, 8 & 11).

For a loss of normal LPSW supply during a Turbine Building Flood, HPSW will provide cooling water to the HPI Pump motor coolers. (Ref. 2, Section 20.1.2).

For both of these events, the source of HPSW inventory is the EWST since power will not be available for Station Blackout and the HPSW Pumps may be rendered inoperable in a Turbine Building flood. No operator action is required because HPSW is supplied via self-contained pressure regulators and/or air-operated fail open valves which open when needed to maintain cooling water supply whenever the normal cooling water supply (LPSW) fails. (Ref. 2, Section 20.1.2).

This SLC does not apply for the tornado scenario. For that scenario, the Auxiliary Service Water (ASW) system supplies backup cooling water to the HPI Pumps since LPSW (primary cooling source) and HPSW (secondary cooling source) are vulnerable to damage by tornado. (Ref. 2, Section 20.1.4.1).

In the Turbine Building Flood event, the SSF provides plant shutdown capabilities. However, HPSW availability does provide additional assurance of being able to mitigate the effects of a Turbine Building flood event. Although the HPSW Pumps are vulnerable to flood damage, the EWST is capable of providing a limited inventory of HPSW. (Ref. 2, Section 20.1.4.7).

REFERENCES

1. 10 CFR 50.65, "Maintenance Rule."
2. OSS-0254.00-00-1002 rev. 4, "Design Basis Specification for HPSW."
3. AP/1,2,3/A/1700/10, "Uncontrollable Flooding of Turbine Building," Approved 9/30/92.
4. Letter dated 4/20/94 from J. W. Hampton (DPC) to NRC regarding supplemental information for revision to CTS 3.4.

5. Selected Licensee Commitments 16.9.1 and 16.9.8.
6. OSC 5945 rev. 0, "HPSW Pump and Fire Protection Test Acceptance Criteria."
7. NRC Safety Evaluation Report, March 10, 1992, Supplemental December 3, 1992 (Station Blackout).
8. UFSAR 8.3.2.2.4, Onsite Power Systems, Station Blackout Analysis.
9. UFSAR 9.2.2.1, Cooling Water Systems, Design Basis.
10. UFSAR 9.2.2.2.2, Cooling Water Systems, Description and Evaluation, HPSW.
11. NRC Safety Evaluation Report, Standby Shutdown Facility, 50-269, -270 & -287.
12. OSC 5771, "PRA Risk-Significant SSC's for the Maintenance Rule."
13. Work Process Manual Section 607, "Maintenance Rule Assessment of Equipment Removed From Service."

16.9 AUXILIARY SYSTEMS

16.9.9 Auxiliary Service Water (ASW) System and Main Steam Atmospheric Dump Valves

COMMITMENT The ASW Pump and the associated piping and valves necessary to supply water to each Unit's Once Through Steam Generators (OTSGs) and to the HPI Pump motors shall be OPERABLE.

---NOTE---
Included as part of the associated valves are the Main Steam Atmospheric Dump Valves.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ASW system inoperable. AND Standby Shutdown Facility (SSF) ASW System is OPERABLE.	A.1 Restore ASW system to OPERABLE status.	30 days
B. ASW system inoperable. AND Standby Shutdown Facility (SSF) ASW System is inoperable.	B.1 Restore ASW system to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Submit report to the NRC outlining plans and procedures to be used to provide for loss of the system.	30 day

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.9.1	Verify appropriate ASW pump discharge capabilities.	92 days
SR 16.9.9.2	Stroke Main Steam Atmospheric Dump Valves.	18 months

BASES

The ASW system is designed to mitigate the consequences of a tornado event by providing emergency feedwater to one or more of the three units at Oconee. This system shall be capable of supplying adequate flow to all units simultaneously to remove core decay heat. The Main Steam atmospheric dump valves are required to be operable for the ASW system to be considered operable because the atmospheric dump valves must be opened to depressurize the OTSGs to allow the low-head ASW pump to supply water to the OTSGs.

Although it is desirable to maintain the ASW system operable to mitigate design basis events, short periods of inoperability are necessary for testing and maintenance to assure a high degree of reliability for the ASW system. Since the probability of any tornado striking the Oconee site is low, a seven day limiting condition for operability (LCO) is reasonable for routine testing and maintenance.

The SSF ASW system is a redundant system and its availability reduces the need of the ASW system. The allowance of 30 days is deemed sufficient time for extended maintenance to be performed on the system as long as the SSF ASW system is available. The testing requirements provide assurance that the minimum OPERABILITY requirements of the ASW system are met.

REFERENCES

1. Design Basis Specification for the Emergency Feedwater and Auxiliary Service Water Systems (OSS-0254.00-00-1000).
2. Oconee UFSAR, Section 9.2.3.
3. Oconee Probabilistic Risk Analysis, Section 3.4
4. Calculational File OSC-2262, "Tornado Protection Analysis."
5. Calculational File OSC-5771, "PRA Risk-Significant SSC's for the Maintenance Rule."
6. Work Process Manual Section 607, "Maintenance Rule Assessment of Equipment Removed From Service."

16.9 AUXILIARY SYSTEMS

16.9.10 Component Cooling (CC) and HPI Seal Injection to Reactor Coolant Pumps

COMMITMENT Both CC flow to Reactor Coolant Pump (RCP) thermal barrier/heat exchangers and High Pressure Injection (HPI) Seal Injection to RCP Seals should be maintained.

APPLICABILITY: Whenever the RCPs are required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------

Station Abnormal Procedures (APs) specify appropriate actions to be taken in the event Component Cooling (CC) system flow and/or HPI seal injection is lost to the RCPs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CC flow to one or more RCP thermal barrier heat exchangers not maintained. <u>OR</u> HPI seal injection to one or more RCP seals not maintained.	A.1 Log unavailability duration in the Operations Log for Maintenance Rule Performance monitoring.	NA
	<u>AND</u>	
	A.2 Perform a Risk Assessment considering equipment out of service.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.9.10.1 NA.	NA

<u>BASES</u>

The intent of this Selected Licensee Commitment is to track Maintenance Rule unavailability and to ensure an acceptable level of risk associated with the removal from service of the following system functions:

1. HPl seal injection to the RCPs.

2. Component Cooling flow to the RCP Thermal Barrier/Heat Exchangers.

In the event that either or both of the listed functions is lost, instructions and limitations with respect to RCP operation are specified in existing plant procedures AP/1,2,3/A/1700/14, AP/1,2,3/A/1700/16, and AP/1,2,3/A/1700/20.

The interactions with other plant systems with respect to plant risk is managed via use of the ONS Risk Assessment Matrix found in WPM-607, in addition to applicable ITS specifications.

Surveillance is by Operations Daily Rounds/Checklists and normal Control Room monitoring.

REFERENCES

1. 10 CFR 50.65, 'Maintenance Rule."
2. OSS-0254.00-00-1001, "Design Basis Specification for the HPl System."
3. OSS-0254.00-00-1022, "Design Basis Specification for the CC System."
4. AP/1,2,3/A/1700/14, revisions 3, 2, 1 respectively
5. AP/1,2,3/A/1700/16, revisions 4. 3, 4 respectively
6. AP/1,2,3/A/1700/20, revisions 2, 2, 2 respectively
7. Calculation No. OSC-5771, "PRA Risk-Significant SSC's for the Maintenance Rule."
8. WPM 607, "Maintenance Rule Assessment of Equipment Removed From Service."

16.9 AUXILIARY SYSTEMS

16.9.11 Turbine Building Flood Protection Measures

COMMITMENT Turbine Building Flood Protection Measures shall be OPERABLE as follows:

a. CCW Pump Discharge Valves (1,2,3CCW-10 through -13) shall be capable of being closed remotely unless one of the following conditions exists:

   1. the unwatering blocks are installed for the associated CCW inlet piping,

   2. the associated condensate coolers CCW flowpath is isolated with locked closed valve(s), the associated waterbox inlet valves are locked closed, the crossover tie valves are locked closed, the CCW inlet piping is vented at the high point to disable the first siphon, and the CCW inlet piping is intact inside the Turbine Building,

   3. Keowee lake level is ≤ 796.5 ft. absolute and the associated CCW inlet piping is vented at the high point to disable the first siphon, or

   4. The CCW Pump Discharge Valve is closed with its breaker open and its handwheel locked.

b. Condenser Outlet Valves (1,2,3CCW-20 through -25) shall be capable of closing automatically when all CCW pumps on the applicable unit are tripped to mitigate certain Turbine Building flood conditions unless one of the following conditions exists:

   1. a condenser outlet valve is closed and air locked with air pressure vented and strongback installed,

   2. a condenser outlet valve is closed with its operator removed and strongback installed,

   3. the unwatering blocks are installed for the associated CCW discharge piping, or

   4. Keowee lake level is ≤ 791 ft. absolute and the associated CCW discharge piping is vented at the high point to prevent reverse siphon flow.

c. Two flowpaths (one each from two different units) shall be available for reverse gravity flow through the Condensate coolers whenever Keowee lake level is greater than 791 ft.

d. Prior to opening any condenser waterbox access hatch or creating any opening in the CCW or LPSW systems > 24 inches diameter (or multiple openings with equivalent diameter > 24 inches) inside the Turbine Building, an isolation boundary with single barriers shall be established to isolate the opening from the lake using the following methods, as applicable:

  1. Any manual valves > 24 inches diameter used for the isolation boundary shall be locked closed,

  2. Any motor-operated valve > 24 inches diameter used for the isolation boundary shall be closed with its breaker locked open and the handwheel locked,

  3. Any condenser outlet valve used for the isolation boundary shall be closed and air-locked with air pressure vented and strongback installed,

  4. A physical barrier, such as unwatering blocks or blank flange, may be used for boundary isolation instead of valves.

e. The Turbine Building/Auxiliary Building boundary wall shall be sealed below Elevation 795 ft. with all water tight doors operable.

f. The Turbine Basement Water Emergency High Level alarm shall be operable.

g. The six foot diameter Turbine Building Flood drain shall be operable.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Turbine Building Flood Protection Measures inoperable.	A.1 ----------NOTE----------- If Turbine Building Flood Protection Measures are inoperable due to planned activities, then these activities shall be performed in a prompt manner without delay. -------------------------	
	Initiate action to restore flood protection measures to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2 ----------NOTE----------- Entry into the associated Condition results in unavailability for all three units. -------------------------	
	Log unavailability duration in the Operations Log for Maintenance Rule Performance monitoring.	None
	<u>AND</u>	
	A.3 Perform a Risk Assessment using the PRA matrix considering CCW integrity not met for all three units.	None

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.11.1	Verify OPERABILITY of Turbine Basement Water Emergency High Level Alarm.	12 months
SR 16.9.11.2	Verify capability to close all four CCW pump discharge valves.	18 months
SR 16.9.11.3	Verify capability to automatically close condenser outlet valves when all CCW pumps are tripped.	18 months

BASES

One of the risk-significant Maintenance Rule functions for the CCW System is to maintain system integrity to prevent or mitigate a Turbine Building flood. The purpose of this Selected Licensee Commitment is to monitor the performance of the major design features associated with this function. To monitor performance of this function, any unavailability must be logged.

The Oconee UFSAR Section 3.4.1.1.1 describes the flood protection measures for the Turbine Building (TB) and Auxiliary Building (AB). These measures are the basis for the commitments in SLC 16.9.11. The flood protection measures were implemented to reduce the overall risk of a Turbine Building flood, as determined by the Oconee Probabilistic Risk Assessment (PRA) study.

Upon detection of a TB flood, operators would trip the CCW pumps which would automatically close all condenser outlet valves. They would also close all CCW pump discharge valves. This may be done using a pushbutton in the control room that closes all four valves on that unit or by closing the valves individually using pushbuttons at the breaker compartment in the Equipment Room. This SLC is intended to ensure that the functional capability of the CCW pump discharge valves and condenser outlet valves will be maintained unless alternative actions have been taken.

In Commitment a, the CCW pump discharge valves shall be capable of being closed remotely. This precludes credit for manual operation of the valves during a flood since there may not be adequate time to take credit for manual operation. Additional options are provided in case the valves cannot be closed remotely. Option 1 requires the unwatering blocks to be installed for the associated CCW inlet piping. Option 2 includes locking closed the

condensate coolers CCW flowpath, the waterbox inlet valves, the crossover tie valves, and venting the CCW inlet piping at the high point. Option 3 involves Keowee lake level ≤ 796.5 feet absolute with the associated CCW inlet piping vented at the high point to disable to disable the first siphon. The high point may be vented by opening valves or by other means, such as manways. Option 4 requires the CCW Pump Discharge Valve to be closed with its breaker open and its handwheel locked. These options provide additional flexibility to allow maintenance to be performed on the CCW pump discharge valves while preventing the possibility of CCW siphoning into the Turbine Building basement.

In Commitment b, additional options are provided to allow maintenance to be performed on the condenser outlet valves. Option 1 allows a condenser outlet valve to be out of service if the valve is blocked closed with the air supply to the valve operator defeated. Option 2 is similar to Option 1 except that it allows the valve operator to be removed for maintenance if the strongback is installed. Option 3 involves installing the unwatering blocks at the CCW discharge and venting the high point of the discharge piping. Option 4 allows the automatic valve operation to be out of service if the lake level is ≤ 791 feet absolute and the high point of the discharge piping is vented. Below this lake level, the CCW discharge pipe could not be refilled from the lake. Venting the high point may be accomplished by opening manways or by any available means. Credit cannot be taken for the normally open mid-point vents on the discharge piping, because these vents may not prevent reverse siphon flow.

Options 3 or 4 of Commitment b will make the affected flowpath incapable of applying towards the requirements in Commitment c, which requires two flowpaths for reverse gravity flow. However, Commitment c may still be met using other available flowpaths (e.g., other units).

For Commitment c, if Keowee lake level is greater than 791 ft., reverse gravity flow can be used to provide suction to the LPSW and SSF ASW pumps. An analysis was performed to determine the optimum flowpath to supply suction to these pumps while minimizing any excess flow that would contribute to additional flooding. This analysis determined that flowpaths through one condensate cooler and one flow control valve on each of two units would be optimum. As a result of this analysis, Condensate Coolers CCW Flow Control Valves for Units 2 and 3 (2, 3CCW-84) have been permanently failed open by having their instrument air supplies removed. If either flowpath through Units 2 or 3 will be unavailable, an alternate flowpath should be provided on Unit 1 by failing open 1CCW-84. A flowpath for reverse gravity flow consists of an open condenser discharge header, one failed-open condensate cooler CCW flow control valve, one open condensate cooler, and an open flowpath to the suction of the LPSW and SSF ASW Pumps.

Commitment d is provided to control activities that would create openings in the CCW or LPSW Systems. These activities are controlled to ensure that such openings are isolated from the lake using physical barriers (e.g., locks) and not just administrative barriers (e.g., valve tags). Per UFSAR Section 3.4.1.1.1, the worst-case flood would involve failure of the expansion joint at the inlet to the condenser. There are other possible failures could lead

to a Turbine Building flood. The flood consequences would vary depending upon the size of the opening and other factors. A flood that involved an opening greater than approximately 24 inches diameter may affect the Low Pressure Service Water (LPSW) pumps. Therefore, emphasis is placed on any activities that would create openings in the piping greater than 24 inches diameter. Commitment d requires that an isolation boundary be established on a case-by-case basis prior to opening a condenser waterbox access-hatch and for any openings > 24 inches, including multiple openings equivalent to 24 inches diameter. Single isolation is acceptable, but the isolation boundary must include physical barriers, such as locked closed valves, and not just administrative barriers, such as valve tags. Physical barriers may include blocks or blank flanges. A stoppel plug or wet-tapping machine may also act as a physical barrier. This SLC is intended to address only the isolation of the opening from the lake. This SLC is applicable to the LPSW pump inlet isolation valves: LPSW-1,-2,-3 and 3LPSW-120,-123. Note that the discharge of each LPSW pump is 18"; thus, there are no valves downstream of the pumps within the scope of the SLC.

If Keowee lake level is greater than 791 ft., reverse gravity flow can be used to provide suction to the LPSW and SSF ASW pumps. An analysis was performed to determine the optimum flowpath to supply suction to these pumps while minimizing any excess flow that would contribute to additional flooding. This analysis determined that flowpaths through one condensate cooler and one flow control valve on each of two units would be optimum. As a result of this analysis, Condensate Coolers CCW Flow Control Valves for Units 2 and 3 (2, 3CCW-84) have been permanently failed open by having their instrument air supplies removed. If either flowpath through Units 2 or 3 will be unavailable, an alternate flowpath should be provided on Unit 1 by failing open 1CCW-84.

REFERENCES

1. UFSAR Sections 3.4.1.1.1, 9.2.2, 9.6, and Figure 9-9, 12/31/97 update.
2. Engineering Directives Manual EDM-210, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants or the Maintenance Rule."
3. OSS-0254.00-00-1003, "Design Basis Specification for the Condenser Circulating.Water (CCW) System," Rev. 11.
4. OSS-0254.00-00-3005, "Design Basis Specification for the Turbine Building Structure," Rev. 1.
5. AP/1,2,3/A/1700/10, "Uncontrollable Flooding of Turbine Building," Approved 4/30/97.
6. Calculation No. C-OSA-SA-83-0002-0, Rev. 0, 3/1/83, "Turbine Building Flood CCW Reverse Flow Analysis."
7. Calculation NO. OSC-6522, Rev. 0, 2/29/96, "Turbine Building Flood CCW Reverse Flow Analysis."
8. Calculation No. OSC-6577, Rev. 0, 6/7/96, "CCW Turbine Building Flood Analysis."
9. PT/1,2,3/A/0261/07, "Dam Failure Test."
10. IP/0/B/0235/03. "Turbine Basement Water Level Alarm System Check."

11. Calculation No. OSC-5771, PRA Risk-Significant SSC's for the Maintenance Rule."
12. Work Process Manual Section 607. "Maintenance Rule Assessment of Equipment Removed From Service".
13. OP/1,2,3/A/1104/12, "Condenser Circulating Water System."
14. Calculation OSC-6081, Rev. 2, CCW Seismic-LOOP Response."
15. Oconee Unit 3 Probabilistic Risk Assessment, Rev. 1, November. 1990.

16.9 AUXILIARY SYSTEMS

16.9.12 Additional Low Pressure Service Water (LPSW) And Siphon Seal Water (SSW) System OPERABILITY Requirements

COMMITMENT The following Structures, Systems and Components (SSCs) shall be OPERABLE:

a. LPSW-4 ("A" LPI COOLER SHELL OUTLET)
b. LPSW-5 ("B" LPI COOLER SHELL OUTLET)
c. LPSW-139 (LPSW SUPPLY TO TB NON-ESSENTIAL HDR)
d. LPSW Pump Minimum Flow Recirculation Lines
e. "A" SSW Header
f. "B" SSW Header

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LPSW flowpath though an LPI cooler isolated by a manual valve.	A.1 Declare associated LPI train inoperable.	Immediately
B. LPSW-4 inoperable and closed. OR LPSW-5 inoperable and closed.	B.1 Declare associated LPI train inoperable.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. LPSW flowpath through an LPI cooler not isolated by manual valve. AND LPSW-4 inoperable and not closed. OR LPSW-5 inoperable and not closed.	C.1 Enter applicable Condition of ITS 3.7.7 for 1 required LPSW pump inoperable.	Immediately
D. One required LPSW pump minimum recirculation line inoperable.	D.1 Enter applicable Condition of ITS 3.7.7 for one required LPSW pump inoperable.	Immediately
E. Two or more Unit 1 and 2 LPSW pump minimum recirculation lines inoperable when three LPSW pumps are required to be OPERABLE by ITS 3.7.7.	E.1 Enter ITS LCO 3.0.3.	Immediately
F. LPSW-139 inoperable.	-------------NOTE------------ Required Action is applicable to each ONS Unit supplied by its associated LPSW System. ---------------------------- F.1 Enter applicable Condition of ITS 3.7.7 for one required LPSW pump inoperable.	 Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. LPSW-139 inoperable on Unit 1. <u>AND</u> LPSW-139 inoperable on Unit 2.	G.1 Enter ITS LCO 3.0.3 for ONS Unit 1 and ONS Unit 2.	Immediately
H. One SSW header inoperable.	H.1 Enter applicable condition of ITS 3.7.8 for one required ECCW Siphon header inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.12.1	Test LPSW-4, LPSW-5, LPSW-139, and check valves in the SSW headers in accordance with the Inservice testing Program.	In accordance with the Inservice Testing Program
SR 16.9.12.2	Test LPSW pump minimum recirculation lines.	18 months

BACKGROUND

The Low Pressure Service Water (LPSW) System provides cooling water for normal and emergency services throughout the station. Safety related functions served by this system include the Reactor Building cooling units (RBCUs), Low Pressure Injection (LPI) coolers, and coolers for the High Pressure Injection (HPI) and Emergency Feedwater (EFW) motors.

The Siphon Seal Water (SSW) System consists of two full capacity headers. The "A" SSW header is supplied by the Unit 1 and 2 LPSW system. The "B" SSW header is supplied by the Unit 3 LPSW System. Each SSW header is capable of providing sealing flow to Unit 1, 2 and 3's ESV and CCW pumps. Both headers are normally in service and aligned to the ESV and CCW pumps. Operability of a SSW header requires that it be supplied from LPSW. Operability of an ESV pump requires that it be aligned to both SSW headers. Each SSW header has a non-safety related HPSW backup. The normal plant configuration is with one SSW header's HPSW supply in service and the other SSW header's HPSW supply isolated. Since the HPSW supply is not safety related, HPSW is not credited to supply the SSW system during a design basis accident.

Operability of both SSW headers is required to mitigate a loss of offsite power (LOOP) and a postulated single failure of a LPSW pump to restart when power is restored. This scenario can result in no LPSW pumps operating on the Unit 3 LPSW System or an inadequate number of operating LPSW pumps on the Unit 1 and 2 LPSW system. SSW flow is unavailable on the affected SSW header. If an ESV pump is operated without seal water, degradation can occur within minutes. Since the standby LPSW pump does not automatically start following a LOOP, Operator action is credited to start the standby LPSW pump in this

scenario. Operator action may not be credited prior to ESV pump degradation. Thus, both SSW headers are required to be operable and aligned to the ESV pumps to prevent interrupting SSW flow. Operability of the SSW headers requires operability of the LPSW System. If a LPSW pump is out of service on the Unit 1 and 2 and/or Unit 3 LPSW System, the SSW supply to the ESV pumps remains single failure proof since both SSW headers are normally in service and aligned to each ESV pump.

APPLICABLE SAFETY ANALYSES

Sufficient LPSW System flow is required to meet the acceptance criteria of containment heat removal safety analyses.

COMMITMENT(S)

The following SSC's shall be OPERABLE:

a. LPSW-4 ("A" LPI COOLER SHELL OUTLET)
b. LPSW-5 ("B" LPI COOLER SHELL OUTLET)
c. LPSW-139 (LPSW SUPPLY TO TB NON-ESSENTIAL HDR)
d. LPSW Pump Minimum Flow Recirculation Lines
e. "A" SSW Header
f. "B" SSW Header

APPLICABILITY

This SLC applies in MODES 1, 2, 3, and 4. This applicability is consistent with the LPSW System operability requirements in Technical Specification 3.7.7. In MODES 5 and 6 the OPERABILITY requirements of the LPSW System are determined by the system it supports.

ACTIONS

<u>A.1, B.1, C.1</u>

During normal operation, LPSW flow is isolated to the LPI coolers with block valves LPSW-4 and LPSW-5 in the closed position. If a LOCA occurs, LPSW-4 and LPSW-5 are required to be opened after Reactor Building Emergency Sump (RBES) recirculation is established. LPSW-251 and LPSW-252 are the normal LPI cooler flow control valves and are normally in AUTO at a setpoint of 3,000 gpm. If a LOCA occurs, Instrument Air (IA) and Auxiliary Instrument Air (AIA) are assumed unavailable since they are not safety related. With LPSW-251 and LPSW-252 failed open and unavailable, LPSW-4 and LPSW-5 are credited for throttling LPI cooler shell side flow to maintain sufficient LPSW pump NPSH

and adequate LPSW flow to the safety related loads. Therefore, as defined in this SLC, LPSW-4 and LPSW-5 operability is met by having the capability to throttle these valves from the control room.

If the LPSW flowpath through an LPI cooler is isolated due to a manual valve, then LPSW pump NPSH and LPSW flow to the other safety related loads would still be adequate. However, the LPSW flow to the affected LPI cooler would not be adequate. Thus, if the LPSW flowpath through an LPI cooler is isolated due to a manual valve, then the affected LPI train shall be declared inoperable.

If LPSW-4 or LPSW-5 is closed and not capable of throttling LPSW flow, then LPSW pump NPSH and LPSW flow to the other safety related loads would still be adequate. However, the LPSW flow to the affected LPI cooler would not be adequate. Thus, if LPSW-4 or LPSW-5 is closed and does not have throttle capability, then the affected LPI train shall be declared inoperable.

If one or both LPSW-4 and LPSW-5 are not closed but not capable of throttling LPSW flow, then LPSW pump NPSH and LPSW flow to the safety related loads may be inadequate. If a single failure of an LPSW pump is not assumed, then sufficient LPSW pump NPSH and LPSW flow to the safety related loads does exist. Thus, if one or both LPSW-4 and LPSW-5 are not closed and do not have throttle capability, then the LPSW system cannot withstand a single failure and the affected unit(s) shall enter the applicable Condition of ITS 3.7.7 for one required LPSW pump inoperable. For Units 1 & 2, both units would be affected if a valve on either unit is inoperable.

D.1, E.1

NSM ON-1,2,33001 removed LPSW-4 and LPSW-5 from ES actuation. By maintaining isolation of LPSW flow to the LPI Coolers during the initial phase of a LOCA, the potential exists for the LPSW pumps to be operated below the manufacturer's recommended minimum continuous flow rate of 4,250 gpm per pump. If all LPSW pumps successfully start and operate during the event, the potential exists for a stronger pump to deadhead a weaker pump during low flow conditions. To avoid damaging a pump due to minimum flow concerns, minimum flow recirculation piping exists for each LPSW pump. The minimum flow recirculation lines ensure the operability of a deadheaded pump until LPSW-4 or LPSW-5 are open on the LOCA unit after RBES recirculation is established. If an LPSW pump's minimum flow recirculation line is inoperable, the LPSW system is not single failure proof and the associated unit shall enter the applicable Condition of ITS 3.7.7 for one required LPSW pump inoperable. If both Unit 3 LPSW pump minimum flow recirculation lines are inoperable, the 72 hour ITS Completion Time is still appropriate because the stronger pump will always have sufficient flow. Likewise, if the Unit 1&2 LPSW system is in a condition that only requires two OPERABLE LPSW pumps per ITS 3.7.7, the minimum flow recirculation lines associated with both OPERABLE pumps may be simultaneously inoperable for a duration of 72 hours permitted by ITS

LCO 3.7.7. If the Unit 1&2 LPSW system is in a condition that requires three OPERABLE LPSW pumps per ITS 3.7.7 and two or more minimum flow recirculation lines are out of service, both Unit 1 and Unit 2 shall enter ITS LCO 3.0.3.

F.1, G.1

During normal operation, valve LPSW-139 is open to supply LPSW flow to the Main Turbine Oil Tank (MTOT) and other various non-essential loads on the applicable unit. In the event of a LOCA, LPSW-139 is credited to close after RBES Recirculation is established, but prior to opening valves LPSW-4 and LPSW-5. Since the Unit 1&2 LPSW system is shared, both LPSW-139 for Unit 1 and LPSW-139 for Unit 2 shall be closed if the non-LOCA unit has tripped due to a concurrent LOOP. Closing LPSW-139 maintains sufficient LPSW pump NPSH and adequate LPSW flow to the safety related loads. Remote closure capability for LPSW-139 shall exist from the control room. If LPSW-139 is not capable of closing, and a single failure of an LPSW pump occurs, LPSW pump NPSH and LPSW flow to the safety related loads may be inadequate. If LPSW-139 for Unit 1 or LPSW-139 for Unit 2 is closed or isolated by system block valves for maintenance, then the valve is still considered operable. Thus, if LPSW-139 is not capable of closing, the associated unit shall enter the applicable Condition of ITS 3.7.7 for one required LPSW pump inoperable. Since the Unit 1&2 LPSW system is shared and LPSW-139 for Units 1 and 2 are normally open, the 72 hour Completion Time applies to both Unit 1 and Unit 2 if LPSW-139 for Unit 1 or LPSW-139 for Unit 2 is inoperable. If both 1LPSW-139 and 2LPSW-139 are inoperable, sufficient LPSW pump NPSH and LPSW flow to the safety related loads may not be available, even without a single failure. This scenario requires both Unit 1 and Unit 2 to enter ITS LCO 3.0.3.

H.1

If a SSW header is inoperable, the ESV system is not single failure proof and the units that require operable ESV systems shall enter the applicable condition of ITS 3.7.8.

SURVEILLANCE REQUIREMENTS

SR 16.9.12.1

This SR requires that LPSW-4, LPSW-5, LPSW-139, and check valves in the SSW headers be tested per Oconee's ASME Section XI IST Program. Testing under this program is adequate to assure operability.

SR 16.9.12.2

This SR requires that the LPSW pump minimum recirculation lines be tested every 18 months. An 18 month frequency is adequate to ensure significant degradation has not occurred due to service water related fouling.

REFERENCES

1. OSS-0254.00-00-1039, Design Basis Specification for the Low Pressure Service Water System, rev. 11.
2. OSC-2280, LPSW Pump NPSH and Minimum Required Lake Level, rev. 10.
3. OSC-4672, Unit 1&2 LPSW System Response to a Large Break LOCA Using a Benchmarked Computer Hydraulic Model, rev. 7.
4. OSC-4489, Predicted Unit 3 LPSW System Response to a Large Break LOCA Using a Benchmarked Computer Hydraulic Model, rev. 5.
5. PT/1/A/0251/023, LPSW System Flow Test, performed on 11/16/97.
6. PT/2/A/0251/023, LPSW System Flow Test, performed on 4/20/96.
7. PT/3/A/0251/023, LPSW System Flow Test, performed on 1/19/97.
8. PT/1,3/A/0251/01, LPSW Pump Test.
9. ITS 3.5.3, 3.7.7 and 3.7.8.
10. Oconee UFSAR Section 9.2.2, 12/31/97 update.
11. Letter from J. W. Hampton, (DPC), to USNRC, dated June 6, 1996, Proposed Technical Specification amendment for LPSW-4, -5.
12. NRC Safety Evaluation Report, dated August 19, 1996, Technical Specification Amendment 217/217/214.
13. PIP 0-098-5871, LOOP With Single Failure Degrades ESV System.

16.9 AUXILIARY SYSTEMS

16.9.13 Spent Fuel Cooling System

COMMITMENT Perform specified SR.

APPLICABILITY: When irradiated fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.13.1	Functionally Test the Spent Fuel Cooling System.	Prior to each refueling

Bases

The requirement(s) of this SLC section were relocated from CTS Table 4.1-2, Item 9 during the conversion to ITS.

Functional testing of the Spent Fuel Cooling System is performed prior to refueling to assure proper system operation.

References

1. UFSAR 9.6.1.

2. DBD OSS-0254.00-00-1006, Rev. 1.

16.9 AUXILIARY SYSTEMS

16.9.14 SSF Diesel Generator Inspection Requirements

COMMITMENT Perform specified SR.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.14.1	Perform Diesel Generator inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.	12 months

Bases

The requirement(s) of this SLC section were relocated from CTS 4.20.3.a.4 during the conversion to Improved Technical Specification.

The testing of the SSF electrical power systems are based upon a review of the surveillance requirements of other similar type of equipment contained within the technical specifications, manufacturer recommendations, and appropriate NRC guidelines.

References

N/A

16.9 AUXILIARY SYSTEMS

16.9.15 Radioactive Material Sources

COMMITMENT Leakage and/or contamination of sealed sources shall be < 0.005 μCi of removable contamination.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Leakage not within limit.	A.1.1 Remove source from use.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.15.1	------------------NOTES------------------ 1. Not required to be met for sources containing ≤ 100μCi of beta and/or gamma-emmitting material or ≤ 10μCi of alpha-emmitting material. 2. Not required to be met for sealed sources that are stored and not being used. 3. Not required to be performed for startup sources subject to core flux. 4. Only required to be met for sealed source containing radioactive material, other than tritium, with a half life greater than 30 days and in any form other than gas. -- Verify leakage and/or contamination of each sealed source is within limit.	 Once each 184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.9.15.2	Verify leakage and/or contamination of each startup source is within limit.	Prior to any repair or maintenance AND Following any repair or maintenance AND Before being initially subjected to core flux

BASES

The requirement(s) of this SLC section were relocated from Technical Specification 4.16 during the conversion to Improved Technical Specification.

This specification assures that leakage from any byproduct, source, and special nuclear radioactive materials sources does not exceed allowable limits.

The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material on the test sample.

Leak testing of sources that are stored and not in use is not required. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date or use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to transfer, sealed sources shall not be put into use until tested.

Sources removed from service due to leakage and/or contamination shall be decontaminated and repaired or disposed of in accordance with NRC regulations.

Leak testing is not required for the check sources contained in extended range area monitors (1)(2)(3)RIA-3, 4, 6, 15, 16, and 17. SR 16.9.15.1 excludes testing sources which are "stored and not being used." Area monitors (1)(2)(3)RIA-3, 4, 6, 15, 16, and 17 have high range detectors which are permanently installed and contain 200μCi keep alive check sources which are not normally accessible and remain sealed within detector housings. Therefore, these sources are considered to be "stored." These sources are considered as "not being used" because they remain stationary within the detector housing and are not subject to any friction due to contact with moving parts.

REFERENCES

N/A

16.9 AUXILIARY SYSTEMS

16.9.16 Reactor Building Polar Crane and Auxiliary Hoist (RCS System Open)

COMMITMENT Operation of the Reactor Building Polar Crane and auxiliary hoist shall be restricted as follows:

a. The Reactor Building Polar Crane shall not be operated over the fuel transfer canal when any fuel assembly is being moved,

b. The auxiliary hoist shall not be operated over the fuel transfer canal when any fuel assembly is being moved,

c. When irradiated fuel is in the reactor building and fuel is not being moved, the reactor building polar crane and auxiliary hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use, and

d. A flagman shall be located at the top of the secondary shield wall when the polar crane hook is above the elevation of the fuel transfer canal when the polar crane is operated in areas away from the fuel transfer canal.

-------------------------NOTE-----------------------------
Commitment part b is not required to be met when the auxiliary hoist is being used to move the fuel assembly.

APPLICABILITY: Fuel in reactor building and reactor vessel head removed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.9.16.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.12.1, 3.12.2, 3.12.3, and 3.12.4 and an associated Technical Specification Interpretation during the conversion to ITS.

Restriction of use of the reactor building polar crane and auxiliary hoist over the fuel transfer canal when the reactor vessel head is removed to those operations necessary for the fuel handling and core internals operations is to preclude the dropping of materials or equipment into the reactor vessel and possibly damaging the fuel to the extent that an escape of fission products would result. The fuel transfer canal will be delineated by readily visible markers at an elevation above which the reactor building polar crane would not normally handle loads.

The fuel transfer canal is the area bounded by the following:

Unit 1
- East and West by the secondary shield walls
- South by the containment wall plate
- North by the 3rd floor handrail

Unit 2
- East and West by the secondary shield walls
- North by the containment wall plate
- South by the 3rd floor handrail .

A fuel assembly is being moved when the Main or Auxiliary bridges are attached to a fuel assembly or a fuel assembly is within the transfer carriage while in the reactor building.

The polar crane is the trolley section, which contains the hooks, blocks and cable drums.

The purpose of having restrictions on use of the polar crane is to prevent the dropping of material or equipment and possibly damaging fuel to the extent that an escape of fission products would occur. The UFSAR, section 9.1.4.1.5 states that the fuel transfer canal is a passageway in the Reactor Building extending from the reactor vessel to the reactor building wall and formed by an upward extension of the primary shield wall. In order to form a boundary for polar crane operation, this area is modified slightly to conform to easily identified structures which provide an extra margin of safety. Therefore the East and West side of the canal are denoted by the secondary shield wall

immediately adjacent to the primary shield wall and the 3rd floor handrail just outside of the primary shield wall of the canal shallow end. The reactor building wall plates determines the final side of the canal area.

It is permissible to operate the polar crane over the fuel transfer canal when absolutely necessary, except during times when any fuel assembly is being moved. Since fuel is in place in the reactor vessel, fuel movement takes place only when fuel is moved into or away from the vessel. Thus once a fuel assembly is attached to the Main or Auxiliary bridge it is considered to be in the act of movement. Also while fuel is in the transfer carriage, being moved into or out of the reactor building, it is considered fuel movement. The reactor building polar crane consists of two connected steel beams on which a trolley assembly moves. The two beams stretch over the full length of the reactor building and are always positioned over the fuel transfer canal. The trolley is the component which actually move equipment up, down and around the reactor building. Therefore for purposes of this SLC, the polar crane consists of the trolley section, whether or not a load is attached.

<u>REFERENCES</u>

N/A

16.9 AUXILIARY SYSTEMS

16.9.17 Reactor Building Polar Crane (RCS at elevated temperature and pressure)

COMMITMENT The Reactor Building Polar Crane shall not be operated over the steam generator compartments.

APPLICABILITY: MODES 1, 2,
MODES 3 and 4 with RCS pressure > 300 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.9.17.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.12.5 and an associated Technical Specification Interpretation during the conversion to ITS.

Restriction of use of the reactor building polar crane over the steam generator compartments during the time when steam could be formed from dropping a load on the steam generator or reactor coolant piping resulting in rupture of the system is required to protect against a loss of coolant accident. The polar crane is the trolley section, which contains the hooks, blocks and cable drums.

The polar crane is the trolley section, which contains the hooks, blocks and cable drums.

The purpose of having restrictions on use of the polar crane is to prevent the dropping of material or equipment and possibly damaging fuel to the extent that an escape of fission products would occur.

The reactor building polar crane consists of two connected steel beams on which a trolley assembly moves. The two beams stretch over the full length of the reactor building and are always positioned over the fuel transfer canal. The trolley is the component which actually move equipment up, down and around the reactor building. Therefore for purposes of this SLC, the polar crane consists of the trolley section, whether or not a load is attached.

References

N/A

16.9 AUXILIARY SYSTEMS

16.9.18 Snubbers

COMMITMENT Hydraulic and Mechanical snubbers specified in the appropriate Station Procedure shall be OPERABLE.

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

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition Entry is allowed for each snubber.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more snubbers inoperable.	A.1 Restore snubber(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met	B.1 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.9.18.1 Perform visual inspections of each snubber in accordance with Table 16.9.18-1.	------NOTE----- The provisions of SLC 16.2.7 do not apply. ------------- In accordance with Table 16.9.18-1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.9.18.2	------------------NOTE------------------ The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information, and the seals shall be replaced so that the maximum expected service life is not exceeded by more than 10% during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Quality Assurance Requirements. -- Verify that the seal service life of hydraulic snubbers is not exceeded by more than 10% between surveillance inspections.	N/A
SR 16.9.18.3	Perform a functional test on a representative sample of hydraulic snubbers and a representative sample of mechanical snubbers in accordance with Table 16.9.18-2.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- In accordance with Table 16.9.18-2

Table 16.9.18-1 (page 1 of 2)
Snubber Visual Inspections

Visual inspections shall verify:

(1) that there are no visible indications of damage or impaired OPERABILITY,

(2) attachments to the foundation or supporting structure are secure, and

(3) in those locations where mechanical snubber movement can be manually induced, the snubbers shall be inspected as follows:

(a) Every 18 months +25%, the inaccessible snubbers shall be inspected near the beginning and the end of the outage.

(b) In the event of a severe dynamic event, snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (i) Manually induced snubber movement, (ii) evaluation of in place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced (or overhauled) before exceeding MODE 5. Re-inspection shall subsequently be performed according to the schedule listed below.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE . However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be tested by starting with the piston at the as found setting and extending the piston rod in the tension mode direction.

Table 16.9.18-1 (page 2 of 2)
Snubber Visual Inspections

All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. Snubber operability will be verified in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	4 months ± 25%
5,6,7	2 months ± 25%
≥8	1 month ± 25%

Note: (1) The required inspection interval shall not be lengthened more than two steps per inspection.

(2) Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility during reactor operation. These two groups may be inspected independently according to be above schedule.

(3) Hydraulic and mechanical snubber inspection schedules are independent.

Table 16.9.18-2 (page 1 of 2)
Snubber Functional Testing

At least once every 18 months +25%, a representative sample, a minimum of 10% of the total of hydraulic and a minimum of 10% of the total mechanical snubbers in use in the plant, shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria, an additional minimum of 10% of the snubbers shall be functionally tested until none are found inoperative or all have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of hydraulic and mechanical snubbers. The representative sample shall be selected randomly from the total population of safety-related hydraulic and mechanical snubbers.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling, and failures shall not require additional testing of other snubbers.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, an engineering evaluation will be performed to determine if the mode of failure could affect other snubbers of the same design. If this is determined, then reporting requirements under 10CFR Part 21 will be examined for applicability.

When a snubber is found inoperable, an engineering evaluation will be performed in accordance with appropriate Station Procedure.

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For hydraulic snubbers specifically required not to displace under continuous load, the ability of the hydraulic snubber to withstand load without displacement shall be verified.

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

Table 16.9.18-2 (page 2 of 2)
Snubber Functional Testing

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression. (Measuring the time required to travel a known distance, under load, is an acceptable method.)

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

BASES

The requirement(s) of this SLC section were relocated from CTS 3.14 and 4.18 during the conversion to ITS.

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup or shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Since the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach MODE 5 will permit an orderly shutdown consistent with standard operating procedures.

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval unless so determined, by the engineer, from a previous window of a schedule. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to

BASES (continued)

determine if any safety-related component or system has been adversely affected by the inoperability of the snubber.

To provide assurance of snubber functional reliability, a representative sample of the installed hydraulic snubbers will be functionally tested every 18 months. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Nuclear Regulatory Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions. Snubbers so exempted shall be listed in a permanent record which references the exemption letter date.

REFERENCES

N/A

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.1 Condensate Inventory Requirements for Emergency Feedwater

COMMITMENT The combined inventory stored in the Upper Surge Tanks (UST) and the Hotwell shall be maintained greater than 145,000 gallons of water.

APPLICABILITY: MODE 1,
MODE 2 with Thermal Power > 2% Rated Thermal Power

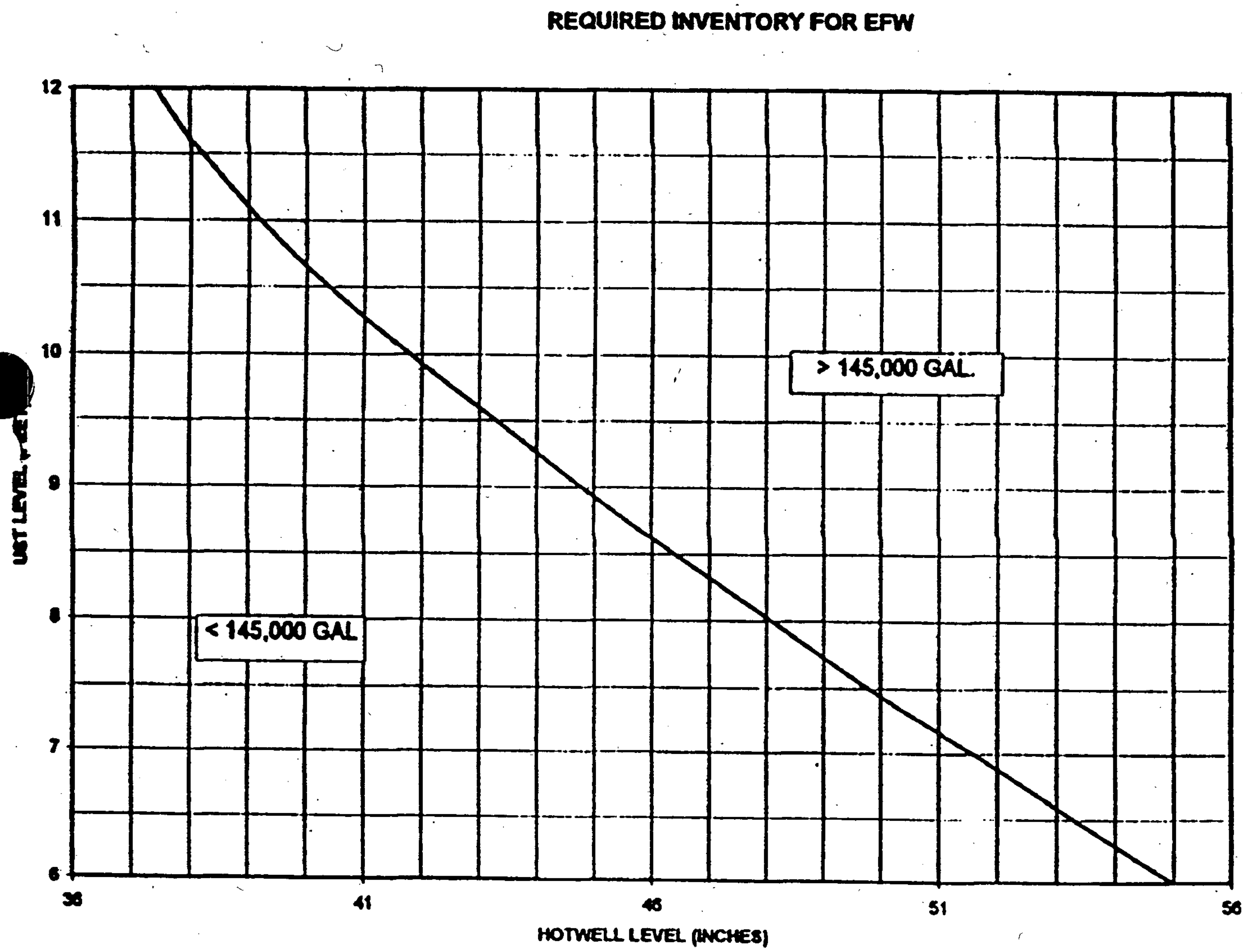
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Combined inventory in the UST and Hotwell ≤ 145,000 gallons of water.	A.1 Restore inventory in the UST and Hotwell to > 145,000 gallons of water.	12 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.10.1.1	Verify combined inventory in the UST and Hotwell is > 145,000 gallons of water (see Figure 16.10.1-1).	6 hours

Figure 16.10.1-1
Required Inventory For EFW



BASES

The EFW design basis requires sufficient water supply be available to cool the Reactor Coolant System, to the point at which the Low Pressure Injection System can provide decay heat removal, after any of the design basis transients for the EFW system (Reference 3). The upper surge tanks are the assured, safety-related water source for the EFW system. The minimum ITS level of 6 feet ensures that adequate time is available to the operator to manually align alternate sources (Reference 2). The Hotwell, which is a non safety-related source, provides this alternate supply of water. Makeup may be available from the condensate storage tanks and the plant demineralized water system, but no credit will be taken for these additional makeup sources in this SLC.

UFSAR Section 10.4.7 states that an inventory of 145,000 gallons of water is required for a 50°F/hr cooldown to the point at which the Low Pressure Injection System can provide decay heat removal. This inventory is based on an initial power level of 102% prior to the loss of main feedwater. The reactor coolant pumps are assumed to be left on to maximize the heat input. This inventory also assumes no recirculation via the turbine bypass valves.

REFERENCES

1. UFSAR Section 10.4.7.1
2. ITS 3.7.6
3. Design Basis Specification for the Emergency Feedwater and the Auxiliary Service Water Systems, Spec. OSS-0254.00-00-1000
4. OSC-5964, EFW Combined Inventory

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.2 Steam Generator Secondary Side Pressure and Temperature (P/T) Limits

COMMITMENT The secondary side of the steam generator shall not be pressurized above 237 psig if the temperature of the vessel shell is below 110°F.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.10.2.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.1.2.5 during the conversion to ITS.

The limitations on steam generator pressure and temperature provides protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure is achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure.

The limitations of 110°F and 237 psig are based on the highest estimated RT_{NDT} of +40°F and the preoperational system hydrostatic test pressure of 1312 psig. The average metal temperature is assumed to be equal to or greater than the coolant temperature. The limitations include margins of 25 psi and 10°F for possible instrument error.

REFERENCES

N/A

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.3 Emergency Feedwater (EFW) Pump and Valve Testing

COMMITMENT Perform specified SRs..

APPLICABILITY: When the associated EFW pump(s) and flow path(s) are required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.10.3.1	Operate turbine-driven and motor-driven EFW pumps on recirculation to the upper surge tank for at least one hour.	In accordance with Inservice Testing Program
SR 16.10.3.2	Verify EFW system flow to each steam generator upon an actual or simulated EFW actuation signal.	Once prior to exceeding 25% RTP after any maintenance or modification which could degrade the EFW flow path

BASES

The requirement(s) of this SLC section were relocated from CTS 4.9.1, 4.9.2 and 4.9.3 during the conversion to ITS.

BASES (continued)

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly. In addition, during operation of the System Flow Test, flow to the steam generators shall be verified by control room indication. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI. The System Flow Test verifies correct total system operation following modifications or repairs.

REFERENCES

UFSAR, Section 10.4.7.4

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.4 Low Presssure Service Water (LPSW) System Testing

COMMITMENT Manually align valves LPSW-4 and LPSW-5 from the control room to demonstrate OPERABILITY of the Low Pressure Injection Coolers.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.10.4.1	Verify valves LPSW-4 and LPSW-5 actuate to the correct position upon manual actuation from the control room.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 4.5.1.1.2.a(2) during the conversion to ITS.

SR 16.10.4.1 verifies that LPSW-4 and -5 (LPSW supply to LPI coolers) respond as required to manual alignment from the control room. The test will be considered satisfactory if valves LPSW-4 and LPSW-5 have completed their travel.

REFERENCES

N/A

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.5 Main Steam Line Break (MSLB) Feedwater Isolation Features

COMMITMENT Perform specified SR.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure ≥ 700 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.10.5.1	Verify main feed pumps, main feedwater control valves and turbine driven emergency feedwater pump are appropriately actuated/inhibited by an actual or simulated MSLB isolation signal.	------NOTE-- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS Table 4.1-2, Item 12 (TSC 96-09) during the conversion to ITS.

SR 16.10.5.1 verifies that equipment operated by the MSLB detection instrumentation functions properly upon reciept of a simulated or actual actuation signal. Main feedwater control valves include both the main feedwater and startup feedwater control valves

REFERENCES

N/A

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.6 Emergency Feedwater Controls

COMMITMENT The controls of the emergency feedwater system shall be independent of the Integrated Control System.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.10.6.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.4.6 during the conversion to ITS.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as sensed by low hydraulic oil pressure. This specific automatic initiation logic is placed in service prior to criticality and may be bypassed when shutdown to present inadvertent actuation during startup and shutdown. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

REFERENCES

N/A

16.10 STEAM AND POWER CONVERSION SYSTEMS

16.10.7 Alternate Source of Emergency Feedwater (EFW)

COMMITMENT

An alternate source of EFW from another unit shall be OPERABLE as follows:

A. The Condensate Storage Tank (CST), Upper Surge Tank (UST) and Hotwell (HW) associated with the credited alternate unit shall be OPERABLE.

AND

B.1 Two motor-driven EFW pumps associated with the credited alternate unit shall be OPERABLE.

OR

B.2 The turbine-driven EFW pump associated with the credited alternate unit shall be OPERABLE.

AND

C. A flow path connecting each required EFW pump from the credited alternate unit to the subject unit's EFW System shall be OPERABLE.

----------------------------NOTE----------------------------

If a unit's EFW system is being utilized to remove decay heat from that unit, then it cannot be utilized to meet the requirements of this SLC.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Alternate source of EFW inoperable.	A.1 ---------NOTE-------- If the alternate source of EFW is inoperable due to planned maintenance, then these activities shall be performed in a prompt manner without delay. --------------------- Initiate action to restore alternate source of EFW to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.10.7.1	--------------------NOTE------------------- This SR may be satisfied by ITS SR 3.7.6.1 for the credited alternate unit. -- Verify combined inventory in the credited alternate unit's UST, CST, and HW is ≥ 72,000 gallons. <u>AND</u> Inventory in the credited alternate unit's UST is ≥ 30,000 gallons.	 12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.10.7.2	--------------------NOTE-------------------- This SR may be satisfied by ITS SR 3.7.5.1 for the credited alternate unit. -- Verify that each EFW manual, and non-automatic power operated valve in the required EFW flow path(s) from the credited alternate unit to the subject unit and, if required, the steam supply flow path to the credited alternate unit's turbine-driven EFW pump that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 16.10.7.3	--------------------NOTE-------------------- This SR may be satisfied by ITS SR 3.7.5.2 for the credited alernate unit. -- Verify the developed head of the credited alternate unit's required EFW pump(s) at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 16.10.7.4	Cycle the required cross-connect valves in the flow path between the credited alternate unit and the subject unit.	In accordance with the Inservice Testing Program
SR 16.10.7.5	--------------------NOTE-------------------- This SR may be satisfied by ITS SR 3.3.14.2 for the credited alternate unit. -- Perform CHANNEL FUNCTIONAL TEST for the manual initiation circuit for the credited alternate unit's EFW pump(s).	92 days

BASES

BACKGROUND

Each unit has an EFW System which is comprised of two motor driven EFW pumps and one turbine driven EFW pump. Each unit's EFW System: 1) normally receives a supply of water from the associated upper surge tanks (UST); 2) can be aligned to the associated hotwell (HW); and 3) has an additional source of water which can be pumped to the USTs (i.e., the associated condensate storage tank (CST)). Each unit's steam turbine driven EFW pump receives steam from either of the associated unit's two main steam headers upstream of the main turbine stop valves (TSVs), or from the Auxiliary Steam System. Each unit's EFW pumps discharge into two distribution headers. Each unit's distribution headers can be cross-connected to a distribution header that is common to all three units. The EFW unit cross-connect valves (i.e., FDW-313 and FDW-314 on each unit) are normally closed. Any EFW pump from the other units can feed either of the steam generators on the subject unit.

For each unit, the steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the associated main steam relief valves (ITS LCO 3.7.1, "Main Steam Relief Valves (MSRVs)"), or atmospheric dump valves. If the main condenser is available, steam may be released via the Turbine Bypass System and recirculated to the HW.

The EFW Systems are described in the UFSAR, Section 10.4.7, (Ref. 1).

APPLICABLE SAFETY ANALYSES

In the event the subject unit's EFW System is rendered unavailable following a High Energy Line Break (HELB) coincident with a single active failure of the subject unit's EFW System, operator action is required to supply feedwater from another unit.

COMMITMENT

This Commitment provides controls to ensure that feedwater can be supplied from another unit's EFW System to mitigate the consequences of an event that renders the subject unit's EFW System unavailable. The EFW System from the credited alternate unit is considered OPERABLE when the components required to provide adequate EFW flow from it to the subject unit's steam generators are OPERABLE.

(continued)

BASES

COMMITMENT (continued)

An alternate source of EFW from another unit shall be OPERABLE as follows:

a. The CST, UST and HW from the credited alternate unit shall be OPERABLE;

b. Two motor-driven EFW pumps or the turbine-driven EFW pump from the credited alternate unit shall be OPERABLE; and

c. One flow path connecting each required EFW pump from the credited alternate unit to the subject unit's EFW System shall be OPERABLE.

If the motor-driven EFW pumps are being credited, then they must be capable of being powered from the offsite power source and the emergency power supply. If the turbine-driven pump is being credited, then its steam supply flow path shall be OPERABLE. Also, the manual initiation circuit for the credited EFW pump(s) must be OPERABLE.

To support the required flow path(s), the cross-connect valves (i.e., FDW-313 and FDW-314) and the EFW flow instrumentation and controls associated with the subject unit are required to be OPERABLE. In addition, the following components from the credited alternate unit are required to be OPERABLE to support the required flow path(s):

a. If the motor-driven EFW pumps are being credited, then cross-connect valves FDW-313 and FDW-314 associated with the credited alternate unit must be OPERABLE; or

b. If the turbine-driven EFW pump is being credited, then only one of the cross-connect valves (i.e., FDW-313 or FDW-314) associated with the credited alternate unit must be OPERABLE.

The OPERABILITY of the credited alternate unit's CST, UST, and HW is determined by maintaining the tank inventory at or above the minimum required inventory.

(continued)

BASES

COMMITMENT (continued)

The Commitment is modified by a Note which states that if a unit's EFW system is being utilized to remove decay heat from that unit, then it cannot be utilized to meet the requirements of this SLC.

APPLICABILITY

In MODES 1, 2, and 3, and MODE 4 when relying upon a steam generator for heat removal for the subject unit, an alternate source of EFW is required to be OPERABLE. These requirements ensure that the alternate source of EFW is capable of mitigating an event that renders the subject unit's EFW system unavailable (i.e., HELB coincident with a single active failure of the subject unit's EFW System). In MODE 4 for the subject unit, the steam generators are used for heat removal unless the DHR System is in operation; the steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.6, "RCS Loops - Mode 4."

In MODES 5 and 6 for the subject unit, the steam generators are not used for DHR. Thus, an alternate unit's EFW System is not required.

ACTIONS

A.1

In the event the alternate source of feedwater is inoperable, action must be initiated to restore the alternate source of EFW to an OPERABLE status immediately. The alternate source of feedwater is comprised of equipment from another unit; thus, its inoperability does not impact the ability of the subject unit's EFW System to perform its function. Typically, while operating in accordance with an Action, an additional single failure is not required to be assumed. Thus, the subject unit's EFW System would remain available to mitigate a HELB. Additional defense-in-depth is provided by the Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW) System. The OPERABILITY requirements for the subject unit's EFW System and the SSF ASW System are contained within ITS 3.7.5 and ITS 3.10.1, respectively.

(continued)

BASES

ACTIONS

A.1 (continued)

This Action is appropriate, because it ensures that action is taken promptly and without delay to restore the ability to supply feedwater to the subject unit via the required equipment of another unit's EFW System.

This Action is modified by a Note which states that if the alternate source of EFW is inoperable due to planned maintenance, then that activity shall be performed in a prompt manner without delay. This means that continuous coverage of the activity should be utilized, and the alternate source of EFW restored to an OPERABLE status as soon as practicable.

SURVEILLANCE REQUIREMENTS

SR 16.10.7.1

This SR verifies that the credited alternate unit's CST, UST, and HW contain the required inventory of cooling water. The 12-hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST, UST, and HW inventory between checks. The 12-hour Frequency is considered adequate in view of other indications in the credited alternate unit's control room, including alarms to alert the operator to abnormal deviations in CST, UST, and HW levels.

This SR is modified by a NOTE which states that ITS SR 3.7.6.1 can be utilized to meet the requirement of this SR. This would occur when the credited alternate unit is being operated in a MODE or condition which requires its CST, UST, and HW to be OPERABLE.

SR 16.10.7.2

Verifying the correct alignment for manual and non-automatic power-operated valves in the required EFW flow path(s) from the credited alternate unit to the subject unit and the required steam supply flow path to the credited alternate unit's turbine-driven EFW pump provides assurance that the proper flow path(s) exist for operation of the required portion of the credited alternate unit's EFW system. This SR does not apply

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 16.10.7.2 (continued)

to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing.

This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31-day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a NOTE which states that ITS SR 3.7.5.1 can be utilized to meet the requirement of this SR. This would occur when the credited alternate unit is being operated in a MODE or condition which requires its EFW System to be OPERABLE.

SR 16.10.7.3

Verifying that the developed head of the credited alternate unit's EFW pump(s) is greater than or equal to the required developed head at the flow test point ensures that EFW pump performance has not degraded below the acceptance criteria during the cycle. Flow and differential head are normal indications of pump performance required by Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test confirms OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 2), at 3 month intervals, satisfies this requirement.

This SR is modified by a NOTE which states that ITS SR 3.7.5.2 can be utilized to meet the requirement of this SR. This would occur when the credited alternate unit is being operated in a MODE or condition which requires its EFW System to be OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 16.10.7.4

This SR verifies that each required cross-over valve in the cross-connect flow path from the credited alternate unit to the subject unit can be manually cycled (FDW-313 and FDW-314 for both the subject and credited alternate unit). Performance of inservice testing in the ASME Code, Section XI (Ref. 2), at 3 month intervals, satisfies this requirement.

SR 16.10.7.5

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the manual initiation circuit for the credited alternate unit's required EFW pump(s). This test verifies that the initiating circuitry is OPERABLE and will actuate the required EFW pump(s) by starting the motor-driven EFW pumps or opening the steam isolation valve that isolates the supply of steam to the drive for the turbine-driven EFW pump. The 92-day Frequency is consistent with the Frequency established in ITS 3.3.14, "Emergency Feedwater (EFW) Pump Initiation Circuitry."

This SR is modified by a NOTE which states that ITS SR 3.3.14.2 can be utilized to meet the requirement of this SR. This would occur when the credited alternate unit is being operated in a MODE or condition which requires the manual initiation circuit for each EFW pump to be OPERABLE.

REFERENCES

1. UFSAR, Section 10.4.7.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.1 Radioactive Liquid Effluents

COMMITMENT Establish conditions for the controlled release of radioactive liquid effluents. Implement the requirements of 10 CFR 20, 10 CFR 50.36a, Appendix A to 10 CFR 50, Appendix I to 10 CFR 50, 40 CFR 141 and 40 CFR 190.

a. Concentration

The concentration of radioactive material released at anytime from the site boundary for liquid effluents to Unrestricted Areas [denoted in Figure 2.1-4(a) of the Oconee Nuclear Station Updated Final Safety Analysis Report] shall be limited to 10 times the effluent concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases the concentration shall be limited to 1×10^{-4} μCi/ml total activity.

b. Dose

The dose or dose commitment to a Member Of The Public from radioactive materials in liquid effluents to Unrestricted Areas shall be limited to:

1. during any calendar quarter:

   $\leq$ 4.5 mrem to the total body

   $\leq$ 15 mrem to any organ, and;

2. during any calendar year:

   $\leq$ 9 mrem to the total body

   $\leq$ 30 mrem to any organ.

c. Liquid Waste Treatment

The appropriate subsystems of the liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid waste prior to their discharge, if the projected dose due to liquid effluent releases to unrestricted areas, when averaged over 31 days would exceed 0.18 mrem to the total body or 0.6 mrem to any organ.

d. Chemical Treatment Ponds (CTP 1 and 2)

1. The quantity of radioactive material in the Chemical Treatment Ponds (CTP) shall be limited so that, for all radionuclides identified, excluding noble gases and tritium, the sum of the ratios of activity (in curies) to the limits in 10 CFR 20, Appendix B, Table 2, column 2 shall not exceed 1.7 x 10^6.

$$\sum_j \frac{Aj}{Cj} < 1.7 \times 10^6$$

Where Aj = pond inventory limit for single radionuclide "j" (curies)

Cj = 10 CFR 20, Appendix B, Table 2, Column 2, concentration for single radionuclide "j" (curies)

2. No powdex resin shall be transferred to the CTPS unless the sum of the activity of the radionuclides identified is less than 0.1% of the limit identified in Commitment d.1.

$$\sum_j \frac{Qj}{Aj} < 1.0 \times 10^{-3}$$

where Qj = radionuclide activity in resin

Aj = pond inventory limit for radionuclide "j"

3. The total radionuclide inventory of used powdex resin transferred to the Chemical Treatment Ponds over the previous 13 weeks, shall not exceed 0.4% of the pond radionuclide inventory limit. Decay of radionuclides may be taken into account in determining inventory levels.

$$Qj_1 + Qj_2 + Qj_3 + \text{------} + Qj_n \leq .004 \times Aj$$

where, Qj = Total inventory of radionuclide j in a transfer

n = Number of transfers to the Chemical Treatment Ponds during the previous 13 - week period.

------------------------NOTE----------------------------
Appendix I dose limits for radioactive liquid effluent releases are applicable only during normal operating conditions which include expected operational occurrences, and are not applicable during unusual operating conditions that result in activation of the Oconee Emergency Plan.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of radioactive material released in liquid effluents to Unrestricted Areas exceeds the limits specified in Commitment a.	A.1 Restore concentration to within the limit.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose from the release of radioactive materials in liquid effluents exceeds any of the limits in Commitment b.	B.1 ---------NOTE---------- Not required during unusual operating conditions that result in activation of the Oconee Emergency Plan, ----------------------	
	Submit report to the regional NRC Office which includes the following: a. Cause(s) for exceeding the limit(s). b. A description of the program of corrective action initiated to: reduce the releases of radioactive materials in liquid effluents, and to keep these levels of radioactive materials in liquid effluents in compliance with the above limits, or as low as reasonably achievable. c. Results of radiological analyses of the drinking water source and the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141.	30 days from the end of the quarter during which the release occurred

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Radioactive liquid waste is discharged without treatment and in excess of the specified limit,	C.1 Submit report to the regional NRC Office which includes the following: a. Cause of equipment or subsystem inoperability. b. Corrective action to restore equipment and prevent recurrence.	30 days
D. Total radioactive inventory of used powdex resins transferred to the Chemical Treatment Ponds over previous 13 weeks greater than 0.4% of the pond radionuclide inventory limit.	D.1 Submit report to the regional NRC Office describing the reason(s) for exceeding the limit and plans for future operation.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.1.1 N/A	N/A

BASES

The concentration commitment is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than 10 times the effluent concentration levels specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its EC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The basic requirements for Selected Licensee Commitments concerning effluent from nuclear power reactors are stated in 10 CFR 50.36a. Compliance with effluent Selected Licensee Commitments will ensure that average annual releases of radioactive material in effluents will be small percentages of the limits specified in the old 10 CFR 20.106 (new 10 CFR 20.1302). The requirements contained in 10 CFR 50.36a further indicate that operational flexibility is allowed, compatible with considerations of health and safety, which may temporarily result in releases higher than such small percentages, but still within the limits specified in the old 10 CFR 20.106 which references Appendix B, Table II concentrations (MPCs). These referenced concentrations are specific values which relate to an annual dose of 500 mrem. It is further indicated in 10 CFR 50.36a that when using operational flexibility, best efforts shall be exerted to keep levels of radioactive materials in effluents as low as reasonably achievable (ALARA) as set forth in 10 CFR 50 Appendix I. Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. Therefore, to accommodate operational flexibility needed for effluent releases, the limits associated with this SLC are based on ten times the instantaneous dose rate value of 50 mrem/year to apply at all times. Compliance with the limits of the new 10 CFR 20.1001 will be demonstrated by operating within the limits of 10 CFR 50, Appendix I, 40 CFR 141 and 40 CFR 190.

Section I of Appendix I of 10 CFR 50 states that this appendix provides specific numerical guides for design objectives and limiting conditions for operation, to assist holders of licenses for light water cooled nuclear power reactors in meeting the requirements to keep releases of radioactive material to unrestricted areas as low as practical and reasonably achievable, during normal reactor operations, including expected operational occurrences. Using the flexibility granted during unusual operating conditions, and the stated applicability of the design objectives for the Oconee Nuclear Station, Appendix I dose limits for radioactive liquid effluent releases are concluded to be not applicable during unusual operating conditions that result in the activation of the Oconee Emergency Plan.

For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This SLC implements the requirements of 10 CFR Part 50.36a. General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix A to 10 CFR Part 50.

The inventory limits of the chemical treatment ponds are based on limiting the consequences of an uncontrolled release of the pond inventory. The short term rate limit (2 mrem/hr) of 10 CFR 20.1301 is applied to 10 CFR 20.1302 in the following expression:

$$\frac{\dfrac{Aj}{1.3 \times 10\ gal} \times 10^{6}\ \dfrac{\mu Ci}{Curie} \times \dfrac{gal}{3785\ ml}}{10 \times Cj} \le \frac{2\ mrem/hr}{500\ mrem/yr} \times \frac{8760\ hr}{yr}$$

$$\frac{Aj}{Cj} \le 1.7 \times 10^{6}$$

Where Aj = pond inventory limit for radionuclide "j" (curies)

Cj = 10 CFR 20. Appendix B. Table 2, Column 2, concentration radionuclide "j"

1.3×10^{6} gal = estimated volume of smaller chemical treatment pond

The transfer limits provide assurance that activity input to the CTP will be minimized.

REFERENCES:

1. 10 CFR Part 20, Appendix B
2. 40 CFR Part 141
3. 10 CFR Part 50, Appendix A and I
4. 40 CFR Part 190
5. Offsite Dose Calculation Manual
6. Regulatory Guide 1.109

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.2 Radioactive Gaseous Effluents

COMMITMENT Establish conditions for the controlled release of radioactive gaseous effluents. Implement the requirements of 10 CFR 20, 10 CFR 50.36a, Appendix A to 10 CFR 50, Appendix I to 10 CFR 50, and 40 CFR 190.

a. Dose Rate

The instantaneous dose rate at the site (exclusion area) boundary for gaseous effluents [Figure 2.1-4(a) of the Oconee Nuclear Station Updated Final Safety Analysis Report due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

1. The dose rate limit for noble gases shall be:

   $\leq$ 500 mrem/yr to the total body

   $\leq$ 3000 mrem/yr to the skin and;

2. The dose rate limit for all radioiodines and for all radioactive materials in particulate form and radionuclides other than noble gases with half-lives greater than 8 days shall be $\leq$ 1500 mrem/yr to any organ.

b. Dose

1. The air dose due to noble gases released in gaseous effluent from the site shall be limited to the following:

   i. During any calendar quarter:

      $\leq$ 15 mrad for gamma radiation

      $\leq$ 30 mrad for beta radiation

   ii. During any calendar year:

      $\leq$ 30 mrad for gamma radiation

      $\leq$ 60 mrad for beta radiation

2. The dose to a Member Of The Public from radioiodines, tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released from the site, shall be limited to the following:

i. During any calendar quarter:

$\leq$ 22.5 mrem to any organ

ii. During any calendar year:

$\leq$ 45 mrem to any organ.

c. Gaseous Radwaste Treatment

1. The Gaseous Radwaste Treatment System shall be used to reduce the noble gases in gaseous wastes prior to their discharge, if the projected gaseous effluent air dose due to gaseous effluent release from the site, when averaged over 31 days exceeds 0.6 mrad for gamma radiation and 1.2 mrad for beta radiation.

2. The Ventilation Treatment Exhaust System shall be used to reduce radioactive materials other than noble gases in gaseous waste prior to their discharge when the projected doses due to effluent releases to unrestricted areas when averaged over 31 days would exceed 0.9 mrem to any organ.

d. Used Oil Incineration

During incineration of used oil contaminated by radioactive material in the Station Auxiliary Boiler, the dose to a Member Of The Public from radioiodines, tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released from the Station Auxiliary Boiler shall be $\leq$ 0.045 mrem to any organ in any calendar year.

---------------------------NOTE---------------------------

The requirement of c.2 does not apply to the Auxiliary Building Exhaust System since it is not "treated" prior to release.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Dose rate exceeds the limits specified in commitment a.	A.1 Restore release rate to within limits.	Immediately
B. Calculated dose exceeds specified limits.	B.1 Submit report to the regional NRC Office which includes the following: a. Cause(s) for exceeding the limit(s), and b. A description of the program of corrective action initiated to: reduce the releases of radioactive materials in gaseous effluents, and to keep these levels of radioactive materials in gaseous effluents in compliance with the specified limits or as low as reasonably achievable.	30 days from the end of the quarter during which the release occurred

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Radioactive gaseous waste is discharged greater than limits specified in Commitment c.1 or c.2. <u>AND</u> Radioactive gaseous waste is discharged without treatment for more than 31 days.	C.1 Submit a report to the regional NRC Office which includes the following: a. Cause of equipment or subsystems inoperability, and b. Corrective action to restore equipment and prevent recurrence.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.2.1 N/A	N/A

BASES

The basic requirements for Selected Licensee Commitments concerning effluent from nuclear power reactors are stated in 10CFR50.36. Compliance with effluent Selected Licensee Commitments will ensure that average annual releases of radioactive material in effluents will be small percentages of the limits specified in the old 10CFR20.106 (new 10CFR20.1302). The requirements contained in 10CFR50.36a further indicate that operational flexibility is allowed, compatible with considerations of health and safety, which may temporarily result in releases higher than such small percentages, but still within the limits specified in the old 10CFR20.106 which references Appendix B. Table II concentrations (MPCs). These referenced concentrations are specific values which relate to an annual dose of 500 mrem to the total body, 3000 mrem to the skin, and 1500 mrem to an infant via the milk animal-milk-infant pathway. It is further indicated in 10CFR50.36a that when using operational flexibility, best efforts shall be exerted to keep levels of radioactive materials in effluents as low is reasonably achievable (ALARA) as set forth in 10CFR50 Appendix I. Therefore, to accommodate operational flexibility needed for effluent releases, the limits associated with gaseous release rate SLCs will be maintained at the current instantaneous dose rate limit for noble gases of 500 mrem/year to the total body and 3000 mrem/year to the skin; and for Iodine-131, for Iodine-133, for tritium. and for all radionuclides in particulate form with half-lives greater than 8 days. an instantaneous dose rate limit of 1500 mrem/year.

The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision I. October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

Equations in the ODCM are provided for determining the actual doses based upon the historical average atmospheric conditions. The release rate commitments for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which are examined in the development Of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides into green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the release of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This commitment implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50.

REFERENCES:

1 10 CFR Part 20. Appendix 8
2. 10 CFR Part 50. Appendix A and I
3. Regulatory Guide 1.109
4. 40 CFR Part 190
5. Offsite Dose Calculation Manual

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.3 Radioactive Effluent Monitoring Instrumentation

COMMITMENT Radioactive Effluent Monitoring Instrumentation shall be OPERABLE as follows:

a. Liquid Effluents

The radioactive liquid effluent monitoring instrumentation channels shown in Table 16.11.3-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of SLC 16.11.1.a are not exceeded.

b. Gaseous Process and Effluents

The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 16.11.3-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of SLC 16.11.2.a are not exceeded.

c. The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded.

------------------------------NOTE------------------------------
Correction to setpoints determined in accordance with Commitment c may be permitted without declaring the channel inoperable.

APPLICABILITY: According to Table 16.11.3-1 and Table 16.11.3-2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Alarm/trip setpoint less conservative than required for one or more Effluent monitoring instrument channels.	A.1 Declare channel inoperable. OR A.2 Suspend release of effluent monitored by the channel.	Immediately Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more required Liquid Effluent monitoring instrument channels inoperable.	B.1	Enter the Condition referenced in Table 16.11.3-1 for the function.	Immediately
	AND		
	B.2	Restore the instrument(s) to OPERABLE status.	30 days
C. One or more required Gaseous Effluent monitoring instrument channels inoperable.	C.1	Enter the Condition referenced in Table 16.11.3-2 for the function.	Immediately
	AND		
	C.2	Restore the instrument(s) to OPERABLE status.	30 days
D. Required Action and associated Completion Time of Required Action B.2 or C.2 not met.	D.1	Explain in next Annual Radiological Effluent Release Report why inoperability was not corrected in a timely manner.	April 30 of following calendar year

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action B.1 and referenced in Table 16.11.3-1. (RIA-33)	E.1.1	Analyze two independent samples in accordance with SLC 16.11.4.	Prior to initiating subsequent release
	<u>AND</u>		
	E.1.2	Conduct two independent data entry checks for release rate calculations	Prior to initiating subsequent release
	<u>AND</u>		
	E.1.3	Conduct two independent valve lineups of the effluent pathway.	Prior to initiating subsequent release
	<u>OR</u>		
	E.2	Suspend release of radioactive effluents by this pathway.	Immediately
F. As required by Required Action B.1 and referenced in Table 16.11.3-1. (RIA-54)	F.1	Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>		
	F.2	Collect and analyze grab samples for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.	Prior to each discrete release of the sump

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action B.1 and referenced in Table 16.11.3-1. (Liquid Radwaste Effluent Line Flow Rate Monitor)	------------NOTE------------- Not required during short, controlled outages of liquid effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ----------------------------	
	G.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	G.2 Estimate flow rate during actual releases.	Immediately <u>AND</u> Once per 4 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action B.1 and referenced in Table 16.11.3-1. (RIA-35, #3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent))	------------NOTE------------- Not required during short, controlled outages of liquid effluent monitoring instrumentation. Short controlled outages are defined as planed removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments. service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ----------------------------	
	H.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	H.2 Collected and analyze grab samples for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.	Immediately <u>AND</u> Once per 12 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action C.1 and referenced in Table 16.11.3-2 for effluent releases from waste gas tanks (RIA-37, RIA-38) or containment purges (RIA-45).	-------------NOTE------------- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments. service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. -----------------------------	
	I.1.1 Analyze two independent samples.	Prior to initiating subsequent release
	AND	
	I.1.2 Conduct two independent data entry checks for release rate calculations	Prior to initiating subsequent release
	AND	
	I.1.3 Conduct two independent valve lineups of the effluent pathway.	Prior to initiating subsequent release
	OR	
	I.2 Suspend release of radioactive effluents by this pathway.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. As required by Required Action C.1 and referenced in Table 16.11.3-2. (Effluent Flow Rate Monitor (Unit Vent , Containment Purge, Interim Radwaste Exhaust, Hot Machine Shop Exhaust, Radwaste Facility Exhaust, Waste Gas Discharge))	----------NOTE--------------- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments. service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ---------------------	
	J.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	J.2 Estimate flow rate	Immediately <u>AND</u> Once per 4 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. As required by Required Action C.1 and referenced in Table 16.11.3-2. (4RIA-45, RIA-53)	------------NOTE------------ Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments. service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ---------------------------	
	K.1 Suspend release of radioactive effluents by this pathway.	Immediately
	OR	
	K.2.1 Collect grab sample.	Immediately AND Once per 8 hours
	AND	
	K.2.2 Analyze grab samples for gross activity (beta and/or gamma).	24 hours from collection of sample

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. As required by Required Action C.1 and referenced in Table 16.11.3-2. (Unit Vent Monitoring Iodine Sampler, Unit Vent Monitoring Particulate Sampler, Interim Radwate Building Ventilation Monitoring Iodine Sampler, Interim Radwate Building Ventilation Monitoring Particulate Sampler, Hot Machine Shop Iodine Sampler, Hot Machine Shop Particulate Sampler, Radwaste Facility Iodine Sampler, Radwaste Facility Particulate Sampler	------------NOTE------------ Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments. service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ----------------------------	
	L.1 Suspend release of radioactive effluents by this pathway.	Immediately
	OR	
	L.2.1 ---------NOTES--------- The collection time of each sample shall not exceed 7 days. --------------------- Collect samples continuously using auxiliary sampling equipment.	Immediately
	AND	
	L.2.2 Analyze each sample.	48 hours from end of each sample collection

CONDITION	REQUIRED ACTION		COMPLETION TIME
M. As required by Required Action C.1 and referenced in Table 16.11.3-2 for effluent releases from ventilation system or condensor air ejectors. (RIA-40)	------------NOTE------------ Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments. service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ---------------------------		
	M.1	Continuously monitor release through the unit vent.	Immediately
	<u>OR</u>		
	M.2	Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>		
	M.3.1	Collect grab sample.	Immediately <u>AND</u> Once per 8 hours
	<u>AND</u>		
	M.3.2	Analyze grab sample for gross activity (beta and/or gamma).	24 hours from collection of grab sample

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.11.3.1	------------------NOTE-------------------- The Channel Response check shall consist of verifying indications during periods of release. Channel response checks shall be made at least once per calendar day on days in which continuous, periodic or batch releases are made. -- Perform Channel Response Check.	During each release via this pathway
SR 16.11.3.2	------------------NOTE-------------------- The Channel Response check shall consist of verifying indications during periods of release. Channel response checks shall be made at least once per calendar day on days in which continuous, periodic or batch releases are made. -- Perform Channel Response Check.	24 hours
SR 16.11.3.3	Perform Source Check.	24 hours
SR 16.11.3.4	Perform Source Check.	31 days
SR 16.11.3.5	Perform Source Check.	92 days

SURVEILLANCE		FREQUENCY
SR 16.11.3.6	------------------NOTE-------------------- The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room annunciation occurs if any of the following conditions exist: 1. Instrument indicates measured levels above the alarm/trip setpoint. 2. Circuit failure (downscale only). -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 16.11.3.7	------------------NOTE-------------------- The Channel Functional Test shall also demonstrate that control room annunciation occurs if any of the following conditions exist: 1. Instrument indicates measured levels above the alarm/trip setpoint. 2. Circuit failure (downscale only). -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 16.11.3.8	Perform CHANNEL FUNCTIONAL TEST.	92 days

SURVEILLANCE		FREQUENCY
SR 16.11.3.9	------------------NOTE--------------------- The initial Channel Calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with the National Institute of Standards and Technology (NIST). The standards shall permit calibrating the system over its intended range of energy and measurement, For subsequent Channel Calibration sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for these requirements.) -- Perform CHANNEL CALIBRATION.	12 months
SR 16.11.3.10	Perform CHANNEL CALIBRATION.	12 months
SR 16.11.3.11	Perform leak test.	When cylinder gates or wicket gates are reworked
SR 16.11.3.12	Perform Source Check.	Within 24 hours prior to each release via associated pathway

Table 16.11.3-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM OPERABLE CHANNELS	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION B.1
1. Monitors Providing Automatic Termination of Release				
a. Liquid Radwaste Effluent Line Monitor, RIA-33	1	At all times	SR 16.11.3.1 SR 16.11.3.3 SR 16.11.3.6 SR 16.11.3.9	E
b. Turbine Building Sump, RIA-54	1	At all times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	F
2. Monitors not Providing Automatic Termination of Release				
Low Pressure Service Water RIA-35	1	At all times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	H
3. Flow Rate Measuring Devices				
a. Liquid Radwaste Effluent Line Flow Rate Monitor (OLW CR0725 or OLW SS0920)	1	At all times	SR 16.11.3.1 SR 16.11.3.10	G
b. Liquid Radwaste Effluent Line Minimum Flow Device	NA	NA	SR 16.11.3.1 SR 16.11.3.10	NA
c. Turbine Building Sump Minimum Flow Device	NA	NA	SR 16.11.3.1 SR 16.11.3.10	NA
d. Low Pressure Service Water Minimum Flow Device	NA	NA	SR 16.11.3.1 SR 16.11.3.10	NA

Table 16.11.3-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

	INSTRUMENT	MINIMUM OPERABLE CHANNELS	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION B.1
e.	Keowee Hydroelectric Tailrace Discharge [(a)]	NA	NA	SR 16.11.3.11	NA
4.	Continuous Composite Sampler				
	#3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent)	1	At all times	SR 16.11.3.2 SR 16.11.3.10	H

(a) Flow is determined from the number of hydro units operating. If no hydro units are operating, leakage flow will be assumed to be 38 cfs based on historical data.

Table 16.11.3-2
GASEOUS EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION C.1
1. Unit Vent Monitoring System				
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Containment Purge Release (RIA-45 - Purge Isolation Function)	1	At All Times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	I
b. Noble Gas Activity Monitor Providing Alarm. (RIA-45 - Vent Stack Monitor Function)	1	At all times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	K
c. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
d. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
e. Effluent Flow Rate Monitor (Unit Vent Flow) (GWD CR0037)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
f. Sampler Flow Rate Monitor [a] (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
g. Effluent Flow Rate Monitor (Containment Purge) (PR CR0082)	1	During Containment Purge Operation	SR 16.11.3.2 SR 16.11.3.10	J
h. CSAE Off Gas Monitor (RIA-40)	1	During Operation of CSAE	SR 16.11.3.2 SR 16.11.3.5 SR 16.11.3.8 SR 16.11.3.9	M
2. Interim Radwaste Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor (RIA - 53)	1	At All Times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	K
b. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
c. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust) (GWD FT0082)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J

Table 16.11.3-2
GASEOUS EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION C.1
e. Sampler Flow Rate Monitor[a] (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
3. Hot Machine Shop Ventilation Sampling System				
a. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
b. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust) (Totalizer)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
d. Sampler Flow Rate Monitor [a] (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
4. Radwaste Facility Ventilation Monitoring System				
a. Noble Gas Activity Monitor (4-RIA-45)	1	At All Times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	K
b. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
c. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
d. Effluent Flow Rate Monitor (Radwaste Facility Exhaust) (OVS CR2060)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
e. Sampler Flow Rate Monitor [a] (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
5. Waste Gas Holdup Tanks				
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RIA-37,-38)[b]	1	During Waste Gas Holdup Tank Releases	SR 16.11.3.1 SR 16.11.3.6 SR 16.11.3.9 SR 16.11.3.12	I
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow) (GWD CR033)	1	During Waste Gas Holdup Tank Releases	SR 16.11.3.1 SR 16.11.3.10	J

(a)Alarms indicating low flow may be substituted for flow measuring devices.
(b)Either Normal or High Range monitor is required dependent upon activity in tank being released.

BASES

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding 10 times the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding applicable dose limits in SLC 16.11.2. The operability end use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

For certain applicable cases, grab samples or flow estimates are required at frequencies between every 4 hours end every 12 hours upon RIA removal from service. SLC 16.11.3 does not explicitly require Action (grab samples or flow estimates) to be initiated immediately upon RIA removal from service, when removal is for the purposes of sample filter changeouts, setpoint adjustments, service checks, or routine maintenance. Therefore, during the defined short, controlled outages, Action is not required.

For the cases in which Action is defined as continuous sampling by auxiliary equipment (Action L) initiation of continuous sampling by auxiliary sampling equipment requires approximately 1 hour. One hour is the accepted reasonable time to initiate collect and change samples. Therefore. for the defined short, controlled outages (not to exceed 1 hour), Action is not required.

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation. the minimum checking frequency stated is deemed adequate.

REFERENCES:

1. 10 CFR Part 20
2. 10 CFR Part 50. Appendix A
3. Offsite Dose Calculation Manual
4. UFSAR, Section 7.2.3.4

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.4 Operational Safety Review

COMMITMENT Required sampling should be performed as detailed in Table 16.11.4-1.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.4.1 N/A	N/A

Table 16.11.4-1
Minimum Sampling Frequency and Analysis Program

Item	Check	Frequency	Lower Limit of Detection (b) of Lab Analysis for Waste
1. Condensate Test Tank, Condensate Monitoring Tank, Laundry-Hot Shower Tank, Waste and Recycle Monitor Tanks	a. Principal Gamma Emitters(c) including Dissolved Noble Gases	Composite Grab Sample prior to release of each batch(h)	<5E-06 μCi/ml (Ce-144 and Mo-99) <5E-07 μCi/ml (Other Gamma Nuclides) <1E-05 μCi/ml (Dissolved Gases) <1E-06 μCi/ml (I-131)
	b. Radiochemical Analysis Sr-89 and Sr-90	Quarterly from all composited batches(f)	<5E-08 μCi/ml
	c. Tritium	Monthly Composite	<1E-05 μCi/ml
	d. Gross Alpha Activity	Monthly Composite	<1E-07 μCi/ml
2. Unit Vent Sampling (Includes Waste Gas Decay Tanks, Reactor Building Purges, Auxiliary Building Ventilation, Spent Fuel Pool Ventilation, Air Ejectors)	a. Iodine Spectrum (a)	Continuous monitor, weekly sample(e)	<1E-10 μCi/cc (I-133) <1E-12 μCi/cc (I-131)
	b. Particulates (a)		
	i. Ce-144 & Mo-99	Weekly Composite(e)	<5E-09 μCi/cc
	ii. Other Principle Gamma Emitters (d)	Weekly Composite(e)	<1E-10 μCi/cc
	iii. Gross Alpha Activity	Monthly, using composite samples of one week	<1E-11 μCi/cc
	iv. Radiochemical Analysis Sr-89, Sr-90	Quarterly Composite	<1E-11 μCi/cc
	c. Gases by Principle Gamma Emitters(d)	Weekly Grab Sample	<1E-04 μCi/cc
	d. Tritium	Weekly Grab Sample	<1E-06 μCi/cc
3. Waste Gas Decay Tank	a. Principle Gamma Emitters(d)	Grab Sample prior to release of each batch	<1E-04 μCi/cc (gases) <1E-10 μCi/cc (particulates and iodines)
	b. Tritium	Grab Sample prior to release of each batch	<1E-06 μCi/cc

Table 16.11.4-1
Minimum Sampling Frequency and Analysis Program

Item		Check		Frequency	Lower Limit of Detection (b) of Lab Analysis for Waste
4.	Reactor Building	a.	Principle Gamma Emitters(d)	Grab sample each purge	<1E-04 μCi/cc (gases) <1E-10 μCi/cc (particulates and iodines)
		b.	Tritium	Grab sample each purge	<1E-06 μCi/cc
5.	Backwash Receiving Tanks		Principle Gamma Emitters including dissolved Noble Gases	Grab Sample prior to release of each batch	NA
6.	#3 Chemical Treatment Pond Effluent[1]	a.	Principle Gamma Emitters(c)	Weekly Continuous Composite(g)	<5E-07 μCi/ml
		b.	I-131	Weekly Continuous Composite(g)	<1E-06 μCi/ml
		c.	Tritium	Monthly Continuous Composite(g)	<1E-05 μCi/ml
		d.	Gross Alpha Activity	Monthly Continuous Composite(g)	<1E-07 μCi/ml
		e.	Sr-89 & Sr-90	Quarterly Continuous Composite(g)	<5E-08 Ci/ml
		f.	Disolved and Entrained gases (Gamma Emitters)	Monthly Grab	<1E-05 Ci/ml
7.	Radwaste Facility Ventilation	a.	Iodine Spectrum(a)	Continuous monitor, weekly sample(e)	(I-133) <1E-09 μCi/cc (I-131) <1E-11 μCi/cc
		b.	Particulate(a)		
		i.	Ce-144 and Mo-99	Weekly Composite(e)	<5E-09 μCi/cc
		ii.	Other Principle Gamma Emitters(d)	Weekly Composite(e)	<1E-10 μCi/cc
		iii.	Gross Alpha Activity	Monthly, using composite samples of one week	<1E-11 μCi/cc
		iv.	Radiochemical Analysis Sr-89, Sr-90	Quarterly Composite	<1E-11 μCi/cc
		c.	Gases by Principle Gamma(d) Emitters	Weekly Grab Sample	<1E-04 μCi/cc
		d.	Tritium	Weekly Grab Sample	<1E-06 μCi/cc

Table 16.11.4-1
Minimum Sampling Frequency and Analysis Program

(a) Samples shall be changed at least once every 24 hours and analysis shall be completed within 48 hours after changing (on or after removal from sampler).

(b) The LLD is defined for purposes of these commitments as the smallest concentration of radioactive material in a sample that would be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation) :

$$LLD = \frac{4.66\ sb}{E \times V \times 2.22E06 \times Y \times \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as micro Curies per unit mass or volume),

sb is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per disintegration),

V is the sample size (in units of mass or volume),

2.22E06 is the number of disintegrations per minute per micro Curie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular nuclide

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples,). NOTE: This assumes decay correction is applied (at the time of analysis) for the duration of sample collection, for the time between collection and analysis, and for the duration of the counting. Additionally, it does not apply to isolated systems such as Waste Gas Decay Tanks and Waste Monitor Tanks.

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not an <u>a posteriori</u> (after the fact) limit for a particular measurement.

(c) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54. Fe-59, Co-58, Co-60. Zn-65, Mo-99, Cs-134, Cs-137. and Ce-141. Ce-144 shall also be meassured, but with a LLD of 5E-06 Ci/ml. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with the above nuclides shall also be analyzed and reported in the Annual Radioactive Effluent Release Report.

(d) The principal gamma emitters for which the LLD commitment applies exclusively are the following radionuclides: Kr-87. Kr-88, Xe-133. Xe-133m, Xe-135. and Xe-138 for gaseous emissions and Mn-54, Fe-59. Co-58, Co-60, Zn-65. Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulates. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides shall also be identified and reported.

(e) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with SLC 16.11.2.a, SLC 16.11.2.b.1, SLC 16.11.2.b.2.

(f) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

(g) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of low of the effluent stream. Prior to analysis, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

Table 16.11.4-1
Minimum Sampling Frequency and Analysis Program

(h) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated, and then thoroughly mixed, to assure representative sampling.

(i) A continuous release is the discharge of liquid wastes of a non-discrete volume, e.g., from a volume of a system that has an input flow durig the continuous release.

BASES

N/A

REFERENCES:

N/A

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.5 Solid Radioactive Waste

COMMITMENT The Solid Radwaste System shall be used in accordance with a Process Control Program, for the solidification of wet radioactive wastes. Prior to the shipment of containers of radioactive wastes from the site, radioactive wastes shall be processed and packaged to ensure meeting the requirements of 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations governing the disposal of radioactive wastes.

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

The Process Control Program shall be used to verify the Solidification of at least one representative test specimens from at least every tenth batch of each type of wet radioactive waste to be solidified.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of 10CFR Part 20 are not satisfied. OR Requirements of 10CFR Part 71 are not satisfied.	A.1	Suspend shipments of defectively packaged solid radioactive wastes from the site.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Any test specimen fails to verify Solidification.	B.1	Suspend solidification of the batch under test until such time as additional test specimens can be obtained, alternative Solidification parameters can be determined in accordance with the Process Control Program, and a subsequent test verifies Solidification. Solidification of the batch may then be resumed using the alternative Solidification parameters determined by the Process Control Program.	Immediately
C. Initial test specimen from a batch of waste fails to verify Solidification.	C.1	Process Control Program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate Solidification. The Process Control Program shall be modified as required to assure Solidification of subsequent batches of waste.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.5.1 NA	NA

BASES

The solid radwaste system will be used whenever radwastes require processing and packaging prior to being shipped offsite. This commitment implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the Process Control Program may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

REFERENCES:

1. 10 CFR Part 50, Appendix A
2. Process Control Program Manual

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.6 Radiological Environmental Monitoring

COMMITMENT

a. The radiological environmental monitoring samples shall be collected in accordance with Table 16.11.6-1 and shall be analyzed pursuant to the requirements of Tables 16.11.6-1, 16.11.6-2 and 16.11.6-3.

b. A land use census shall be conducted and shall identify the location of the nearest milk animal and the nearest residence in each of the 16 meteorological sectors within a distance of five miles. Broad leaf vegetation sampling shall be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

c. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program. A summary of the results obtained as part of the Interlaboratory Comparison Program and in accordance with the methodology and parameters in the ODCM shall be included in the Annual Radiological Environmental Operating Report.

d. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

---NOTE---

If samples required by Commitment part a, become permanently unavailable from any of the required sample locations, the locations from which samples were unavailable may then be deleted from the program provided replacement samples were obtained and added to the environmental monitoring program, if available. These new locations will be identified in the Annual Radioactive Effluent Release report.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Radiological environmental monitoring program is not conducted as required.	A.1	Submit a description of the reason for not conducting the program as required and plans to prevent a recurrence shall be included in the Annual Radiological Environmental Operating Report.	May 15 of following calendar year
B. Land use census identifies a Location which yields a calculated dose or dose commitment (via the same exposure pathway) greater than a location from which samples are currently being obtained.	B.1	---------NOTE--------- The sampling location having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. --------------------- Add new location to the radiological environmental monitoring program.	30 days
	<u>AND</u>		
	B.2	Identify new locations in the next Annual Radioactive Effluent Release report.	April 30 of following calendar year

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Interlaboratory Comparison Program analyses not performed as required.	C.1	Report corrective actions in the Annual Radiological Environmental Operating Report.	May 15 of following calendar year

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.11.6.1	Conduct land use census during growing season using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.	12 months

Table 16.11.6-1
Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample Locations (b)	Sampling and Collection Frequency (d)	Time and Frequency of Analysis
1. AIRBORNE			
Radioiodine and Particulates	5	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine canister: I-131 analysis weekly. Particulate sampler: Gross beta radioactivity analysis following filter change; and gamma isotopic analysis of composite (by location) quarterly. (c)
2. DIRECT RADIATION	40	Quarterly.	Gamma dose quarterly.
3. WATERBORNE			
a. Surface	2	Composite (a) sample over a 1-month period.	Gamma isotopic analysis monthly. Composite for tritium analysis quarterly.
b. Drinking	3	Composite (a) sample over a 1-month period.	Composite for gross beta and gamma isotopic analyses monthly. Composite for tritium analysis quarterly.
c. Sediment from Shoreline	2	Semiannually.	Gamma isotopic analysis semiannually.

Table 16.11.6-1
Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample Locations (b)	Sampling and Collection Frequency (d)	Time and Frequency of Analysis
4. INGESTION			
a. Milk	3	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
b. Fish	2	Semiannually. One sample of each of the following species: 1. Bass 2. Catfish	Gamma isotopic analysis semiannually on edible portion.
c. Broad-leaf Vegetation	2	Monthly.	Gamma isotopic analysis monthly.

(a) Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.

(b) Sample locations are identified in the ODCM.

(c) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(d) Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

Table 16.11.6-2
Maximum Values for the Lower Limits of Detection (LLD) (a)(c)

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Broad-leaf Vegetation (pCi/kg, wet)	Sediment (pCi/kg, dry)
Gross Beta	4	1E-02				
H_3	2,000					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			
Co-60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	15(b)	7E-02		1	60	
Cs-134	15	5E-02	130	15	60	150
Cs-137	18	6E-02	150	18	80	180
Ba-140	60			60		
La-140	15			15		

(a) The LLD is defined, for purposes of these commitments, as the smallest concentration of radioactive material in a sample with 95% probability of detection and with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ Sb}{E \times V \times 2.22 \times Y \times \exp(-\lambda\Delta t)}$$

Where:

Table 16.11.6-2
Maximum Values for the Lower Limits of Detection (LLD) (a)(c)

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume)

Sb is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per disintegration)

V is the sample size (in units of mass or volume)

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting

Typical values of E, V, Y and Δt should be used in the calculation.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances, may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

(b) LLD for gamma isotopic analysis for I-131 in drinking water samples. Low level I-131 analysis on drinking water will not be routinely performed because the calculated dose from I-131 in drinking water at all locations is less than 1 mrem per year. Low level I-131 analyses will be performed if abnormal releases occur which could reasonably result in > 1 pCi/liter of I-131 in drinking water. For low level analyses of I-131 an LLD of 1 pCi/liter will be achieved.

(c) Other peaks which are measurable and identifiable, together with the radionuclides in Table 16.11.6-2, shall be identified and reported.

Table 16.11.6-3
Reporting Levels for Radioactivity Concentrations in Environmental Samples

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Broad-leaf Vegetation (pCi/kg, wet)
H-3	2E04(a)				
Mn-54	1E03		3E04		
Fe-59	4E02		1E04		
Co-58	1E03		3E04		
Co-60	3E02		1E04		
Zn-65	3E02		2E04		
Zr-Nb-95	4E02				
I-131	2(b)	1.0		3	1E02
Cs-134	30	10	1E03	60	1E03
Cs-137	50	20	2E03	70	2E03
Ba-La-140	2E02			3E02	

(a) For drinking water samples. This is 40 CFR Part 141 value.

(b) If low level I-131 analyses are performed.

BASES

The environmental monitoring program required by this commitment provides measurements of radiation and of radioactive materials in those exposure pathways and for whose radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 16.11.6-2 are considered optimum for routine environmental measurements in industrial laboratories. The specified lower limits of detection correspond to less than the 10 CFR 50. Appendix I, design objective dose-equivalent of 45 mrem/year for atmospheric releases to the most sensitive organ and individual. The land use census commitment is provided to assure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are provided if required by the results of this census.

The requirements for participation in an Interlaboratory Comparison Program is provided to assure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

The following requirement(s) were relocated from the CTS 6.4.4.f during the conversion to ITS.

The station shall have a program to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in UFSAR Chapter 16, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census; and,

3. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of

radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

<u>REFERENCES</u>:

1. 10 CFR Part 50. Appendix I
2. Offsite Dose Calculation Manual

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.7 Dose Calculations

COMMITMENT The annual (calendar year) dose or dose commitment, to any Member of The Public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to ≤ 25 mrems to the total body or to any organ, except the thyroid, which shall be limited to ≤ 75 mrems.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of SLC 16.11.1.b, SLC 16.11.2.b.1, or SLC 16.11.2.b.2	A.1 Determine by calculation, including direct radiation contributions from the reactor units and from outside storage tanks, whether the limits of Commitment 16.11.7 have been exceeded.	None

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose exceeds limits of Commitment 16.11.7.	--------------NOTE----------- This Special Report, as defined in 10 CFR Part 20.2203(a), shall include an analysis that estimates the radiation exposure (dose) to a Member Of The Public from uranium fuel cycle sources, (including all effluent pathways and direct radiation), for the calendar year that includes the release(s) covered by this report. It shall also describe the levels of radiation and concentration of radioactive material involved, and the cause of the exposure levels or concentrations. -----------------------------	
	B.1 Prepare and submit to the Commission a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the specified limits and includes the schedule for achieving conformance with the specified limits.	30 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Calculated dose exceeds limit of Commitment 16.11.7. AND Release condition resulting in violation of 40 CFR 190 not corrected at time of report submittal.	C.1 ---------NOTE--------- Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. ---------------------- Include a request for a variance in accordance with the provisions of 40 CFR Part 190.	30 days from exceeding the limit

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.11.7.1	Determine cumulative dose contributions from liquid effluents in accordance with Offsite Dose Calculation Manual.	31 days
SR 16.11.7.2	Determine cumulative dose contributions from gaseous effluents in accordance with Offsite Dose Calculation Manual.	31 days

BASES

The dose commitment is provided to assure that the release of radioactive material in liquid and gaseous effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I in that conformance with the guides of Appendix I is to be shown by calculations and procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated.

that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated.

REFERENCES:

1. 10 CFR Part 20
2. 40 CFR Part 190
3. Offsite Dose Calculation Manual
4. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.8 Reports

COMMITMENT Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable SLC:

a. Radioactive Liquid Effluents,
   Dose, SLC 16.11.1.b
   Liquid Waste Treatment, SLC 16.11.1.c
   Chemical Treatment Ponds, SLC 16.11.1.d

b. Radioactive Gaseous Effluents.
   Dose, SLC 16.11.2.b
   Gaseous Radwaste Treatment, SLC 16.11.2.c

c. Radiological Environmental Monitoring Program, SLC 16.11.6.a, b, and c

d. Land Use Census, SLC 16.11.6.d

e. Dose Calculations, SLC 16.11.7

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Individual milk samples show I-131 concentrations of 10 picocuries per liter or greater.	A.1	Submit plan advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.	7 days

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Milk samples collected over a calendar quarter show average concentrations of ≥ 4.8 picoCuries per liter	B.1	Submit a plan advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.8.1 NA	NA

BASES

Reference applicable commitments.

REFERENCES:

1. 10 CFR Part 20
2. 40 CFR Part 190
3. Offsite Dose Calculation Manual

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.9 Radioactive Effluent Release Report

COMMITMENT The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year.

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

The Annual Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the station during the reporting period.

The annual Radioactive Effluent Release Report shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter.

The Annual Radioactive Effluent Release Report shall include an assessment of the radiation dose from radioactive effluents to members of the public due to their activities inside the unrestricted area boundary during the reporting period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports.

The Annual Radioactive Effluent Release Report shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved;

b. Cause(s) for the unplanned release;

c. Actions taken to prevent recurrence; and,

d. Consequences of the unplanned release.

The Annual Radioactive Effluent Release Report shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the station during each calendar quarter. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The annual average meteorological conditions shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment

of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual.

The Annual Radioactive Effluent Release Report shall include an explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation out of service for greater than 30 days was not corrected in a timely manner per SLC 16.11.3.

The Annual Radioactive Effluent Release Report shall include the following information for each type of solid waste shipped offsite during the report period:

a. Total container volume (cubic meters);

b. Total curie quantity (determined by measurement or estimate);

c. Principal radionuclides (determined by measurement or estimate);

d. Type of waste, (e.g., spent resin. compacted dry waste evaporator bottoms);

e. Number of shipments; and,

f. Solidification agent (e.g., cement, or other approved agents (media)).

The Annual Radioactive Effluent Release Report shall include a list and description of unplanned releases from the site to Unrestricted Areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Annual Radioactive Effluent Release Report shall include any changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census.

The Annual Radioactive Effluent Release Report shall also include an assessment of radiation doses to the likely most exposed Member Of The Public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Methods for calculating the dose contribution from liquid and gaseous effluents are given in the ODCM.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.9.1 NA	NA

BASES

NA

REFERENCES:

1. Oconee ITS
2. Offsite Dose Calculation Manual

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.10 Radiological Environmental Operating Report

COMMITMENT Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 15 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations. and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses. If harmful effects are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environment Operating Report shall include a summary of the results obtained as part of the required Interlaboratory Comparison Program and in accordance with the ODCM. Alternatively, participants in the EPA cross-check program shall provide the EPA program code designation for the unit.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the radiological environmental samples required by SLCs taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as practical in a supplementary report.

The initial report shall also include the following: a summary description of the radiological environmental monitoring program including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and, the result of land use censuses. Subsequent reports shall describe all substantial changes in these aspects.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.10.1 NA	NA

BASES

NA

REFERENCES:

1. Oconee ITS
2. Offsite Dose Calculation Manual

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.11 Iodine Radiation Monitoring Filters

COMMITMENT Assure that the iodine radiation monitoring filters perform their intended function.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.11.1 Remove and replace iodine radiation monitoring filters in RIA-44.	30 days of operation
SR 16.11.11.2 Discard spare iodine radiation monitoring filters.	After 24 months of shelf life.

BASES

The purpose of this commitment is to assure the reliability of the iodine radiation monitoring charcoal filters.

REFERENCES:

1. Oconee CTS Amendment No. 3/3 SER date July, 1974.

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.12 Radioactive Material in Outside Temporary Tanks Exceeding Limit

COMMITMENT The quantity of radioactive material in outside temporary storage tanks shall not exceed the limit specified in ITS 5.5.13.c.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The quantity of radioactive material in outside temporary storage tank not within limit.	A.1 Suspend addition of radioactive material to tank.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.11.12.1	Verify the quantity of radioactive material contained in each of the outside temporary tanks is within the limit by analyzing a representative sample of the tanks' contents.	within 7 days after addition of radioactive materials to the tank

BASES

The requirement(s) of this SLC section were relocated from CTS 3.9.1.c during the conversion to ITS.

The tanks included in this specification are all those outdoor radwaste liquid storage tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system. Restricting the quantify of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of a tank's contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

REFERENCES

N/A

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.13 Radioactive Material in Waste Gas Holdup Tank Exceeding Limit

COMMITMENT The quantity of radioactive material in the Waste Gas Holdup tanks shall not exceed the limit specified in ITS 5.5.13.b.

APPLICABILITY: At all times.

ACTIONS

-----NOTE-----
Separate Condition Entry is allowed for each tank.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The quantity of radioactive material in the Waste Gas Holdup tank not within limit.	A.1	Suspend addition of radioactive material to tank.	Immediately
	AND		
	A.2	Reduce tank contents to within limit.	48 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.11.13.1	Verify quantity of radioactive materials in each tank is within limit.	24 hours when tank is being filled

BASES

The requirement(s) of this SLC section were relocated from CTS 3.10.1.b and 3.10.1.c during the conversion to ITS.

Restricting the quantity of radioactivity contained in each waste gas holdup tank provides assurance that in the event of an uncontrolled release of the tank contents, the resulting total body exposure to an individual at the exclusion area boundary will not exceed 0.5 rem.

REFERENCE

UFSAR, Section 15.10

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.14 Explosive Gas Mixture

COMMITMENT The concentration of Hydrogen in the Waste Gas Holdup Tanks shall be ≤ 3% by volume.

APPLICABILITY: At all times.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition Entry is allowed for each tank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of Hydrogen in Waste Gas Holdup tank is > 3% and ≤ 4% by volume.	A.1 Reduce Concentration of Hydrogen to within limit.	48 hours
B. Concentration of Hydrogen in Waste Gas Holdup tank is > 4% by volume.	B.1 Suspend addition of waste gases to tank.	Immediately
	AND	
	B.2 Reduce Concentration of Hydrogen to within limit.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.11.14.1 Verify Hydrogen concentration in Waste Gas Holdup Tank is ≤ 3% by volume.	5 times/week on each tank when in service AND once within 24 hours after isolation of the tank

BASES

The requirement(s) of this SLC section were relocated from CTS 3.10.2 and Table 4.1-3, Item 13 during the conversion to ITS.

This Commitment is provided to ensure that the concentration of potentially explosive gas mixtures contained in the Waste Gas Holdup Tanks is maintained below the flammability limits of hydrogen. (Administrative controls are used to prevent the hydrogen concentrations from reaching the flammability limit.) These controls include sampling each tank 5 times a week while in service, and/or once in 24 hours after isolation of the tank; injection of dilutants to reduce the concentration of hydrogen below its flammability limits provides assurance that the releases of radioactive material will be controlled in conformance with the requirements of GDC 60 of Appendix A to CFR Part 50.

REFERENCES

N/A

16.12 REFUELING OPERATIONS

16.12.1 Decay Time for Movement of Irradiated Fuel

COMMITMENT Irradiated fuel shall not be moved from the reactor until the unit has been subcritical for at least 72 hours

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.12.1.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.8.11 during the conversion to ITS.

The safety analysis for the fuel handling accident is based on the assumption that the reactor has been shutdown for 72 hours.

REFERENCES

UFSAR, Section 15.11.2.1

16.12 REFUELING OPERATIONS

16.12.2 Area Radiation Monitoring for Fuel Loading and Refueling

COMMITMENT Radiation levels in the reactor building refueling area shall be monitored by RIA-3 and by a portable bridge monitor for each bridge which is being used for fuel handling. Radiation levels in the spent fuel storage area shall be monitored by RIA-6 and by a portable bridge monitor.

APPLICABILITY: During fuel handling.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required Area Radiation Monitor not monitoring radiation levels.	A.1	Use portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.12.2.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.8.1 during the conversion to ITS.

Continuous monitoring of radiation levels provides immediate indication of an unsafe condition.

REFERENCES

N/A

16.12 REFUELING OPERATIONS

16.12.3 Communication Between Control Room and Refueling Personnel

COMMITMENT Direct communications between the control room and the refueling personnel in the reactor building shall exist.

APPLICABILITY: During CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.12.3.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.8.5 during the conversion to ITS.

This Commitment allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

REFERENCES

N/A

16.12 REFUELING OPERATIONS

16.12.4 Handling of Irradiated Fuel Assemblies

COMMITMENT When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

---------------------------NOTE---------------------------
The Auxiliary Hoist may be used provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

APPLICABILITY: During movement of irradiated fuel assemblies inside containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.12.4.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.8.8 during the conversion to ITS.

REFERENCES

N/A

16.12 REFUELING OPERATIONS

16.12.5 Loads Suspended Over Spent Fuel in Spent Fuel Pool

COMMITMENT No suspended loads of more than 3000 lb_m shall be transported over spent fuel stored in the spent fuel pool.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.12.5.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.8.14 during the conversion to ITS.

This commitment is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.1 Fire Brigade

COMMITMENT A Fire Brigade of five members shall be maintained onsite at all times.

----------------------------NOTE--------------------------------
This excludes 3 members of the minimum operating shift requirements that are required to be present in the control rooms.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.1.1 NA	NA

BASES

The primary purpose of the Fire Protection Program is to minimize both the probability and consequences of postulated fires. Despite designed active and passive Fire Protection Systems installed throughout the plant, a properly trained and equipped Fire Brigade organization of at least five members is needed to provide immediate response to fires that may occur at the site.

Fire Brigade equipment and training conform to Oconee's commitments to Appendix A to Branch Technical Position 9.5-1 and supplemental NRC Staff guidelines including Nuclear Plant Functional Responsibilities, Administrative Controls and Quality Assurance.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License Conditions.

The following requirement was relocated from the CTS 6.1.1.8 during the conversion to ITS.

A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

REFERENCES:

1. Oconee UFSAR, Chapter 9.5.1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, as revised.
4. Duke letter of January 16, 1978 to NRC in response to "Nuclear Plant Functional Responsibilities, Administrative Controls, and Quality Assurance".

16.13 CONDUCT OF OPERATIONS

16.13.2 Technical Review and Control

COMMITMENT A Technical Review and Control Program covering the preparation, review, and approval of documents important to station operation shall be established and maintained for the site.

Personnel performing the preparation, review, and approval activities covered by this commitment shall meet or exceed the qualifications of ANSI N18.1-1971 (the conformance status for this standard is as listed in Table 17-1 of the Duke Power Topical Report, Quality Assurance Program, Duke-1-A).

a. The preparation, review, and approval of station procedures shall be done in accordance with station Technical Specifications. Individuals responsible for these reviews shall be members of the supervisory staff assigned to the site, be previously designated by the Site Vice President as a Qualified Reviewer, and successfully complete the site Qualified Reviewer training program. Review of environmental radiological analysis procedures, shall be performed by the General Manager, Environmental Services or a designee. Each such review shall include a determination of whether or not additional, cross-disciplinary review shall be performed by the appropriately designated site review personnel.

b. Proposed modifications shall be designed and the design reviewed in accordance with station Technical Specifications. The proposed modification design, the design review, and design approval shall be in accordance with ANSI N45.2.11 as described in Table 17-1 of the Duke Power Topical Report, Quality Assurance Program, Duke 1-A. Proposed modifications to nuclear safety related structures, systems, and components shall be approved prior to implementation by the Station Manager or the Manager of Engineering; or for the Station Manager, by a Maintenance Superintendent, the Operations Superintendent, or the Work Control Superintendent, as previously designated by the Station Manager. Upon implementation approval, the modification shall be implemented in accordance with the Duke Power Nuclear Station Modification Program and approved procedures (as discussed in Item a above).

c. Proposed changes to the station Technical Specifications shall be prepared in accordance with station Technical Specifications. Each proposed Technical Specification change shall be reviewed by the Plant Operations Review Committee (PORC) and the Nuclear Safety Review Board (NSRB) prior to submittal to the Nuclear Regulatory Commission. Proposed changes to the Technical Specifications shall be

approved by the Station Manager, or for the Station Manager by a designated manager or company officer. Technical Specifications submittal cover letters shall be signed by an officer of Duke Power Company.

d. Proposed tests and experiments which affect station nuclear safety and are not addressed in the UFSAR or Technical Specifications shall be reviewed by the Plant Operations Review Committee (PORC).

e. Incidents reportable pursuant to station Technical Specifications and all violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Manager, Safety Assurance and provided to the Site Vice President and the Plant Operations Review Committee (PORC).

f. The Manager, Safety Assurance shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Site Vice President. Such reports shall be provided to the Plant Operations Review Committee (PORC).

g. The Manager, Safety Assurance shall assure the performance of a review by a knowledgeable individual/organization of every unplanned onsite release of radioactive material to the environs, including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence to the Site Vice President. and to the Plant Operations Review Committee (PORC).

h. The Manager, Safety Assurance shall assure the performance of a review by a knowledgeable individual/organization of changes to the Process Control Program, Offsite Dose Calculation Manual (ODCM), and Radwaste Treatment Systems.

i. The Manager, Safety Assurance shall ensure the performance of a review by a knowledgeable individual/organization of the Fire Protection program and implementing procedures and submittal of recommended changes to the Director, Organization Effectiveness Services.

j. Reports documenting each of the activities performed under this commitment shall be maintained. Copies shall be provided to the NSRB.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.2.1 NA	NA

BASES

The requirements contained in this selected licensee commitment were relocated from the Oconee Technical Specifications with the approval of the U. S. Nuclear Regulatory Commission. Changes to this SLC shall be considered a change in an NRC commitment and shall be made only in accordance with the approved Compliance Manual Procedure for the Control of Selected Licensee Commitments and by use of the 10 CFR 50.59 evaluation process.

This SLC implements the review requirements of ANSI N18.7-1976/ANS-3.2 and ANSI N45.2.11-1974 as referenced in the Duke Power Company Topical Report, Quality Assurance Program, Duke-1-A.

REFERENCES:

1. ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel
2. ANSI N18.7-1976/ANS-3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
3. ANSI N45.2.11-1974, Quality Assurance ReqUirementS for the Design of Nuclear Power Plants
4. Compliance Manual Procedure for the Control of Selected Licensee Commitments
5. Nuclear System Directive 209, 10 CFR 50.59 Evaluations
6. 10 CFR 50.59
7. Nuclear System Directive 703, Administrative Instructions for Station Procedures

16.13 CONDUCT OF OPERATIONS

16.13.3 Plant Operations Review Committee

COMMITMENT A Plant Operations Review Committee (PORC) shall be established and maintained for the site. The PORC shall be composed of the Manager of Safety Assurance, the Station Manager and his/her direct reports most responsible for station operation and maintenance, the Manager of Engineering and his/her direct reports most responsible for engineering support of station operation and maintenance, or designated alternates. The PORC Chairperson, members, and alternate members shall be qualified in accordance with ANSI N18.1-1971 and be appointed by the Site Vice President. The quorum necessary for conducting the PORC functions shall consist of the Chairperson, or his/her designated alternate, and at least three other PORC members including alternates.

Reports of reviews encompassed by this Selected Licensee Commitment shall be prepared and forwarded to the Site Vice President and the Nuclear Safety Review Board.

a. The PORC shall be responsible for reviewing the following prior to final approval:

   1. All proposed tests and experiments which affect station nuclear safety and are not addressed in the UFSAR or Technical Specifications;

   2. OPERABILITY evaluations resulting in a Justification for Continued Operation and a proposal for discretionary enforcement;

   3. OPERABILITY evaluations resulting in the decision that affected systems, structure or components are OPERABLE but degraded; and

   4. All proposed changes to the station Technical Specifications, Bases, or Facility Operating License.

b. The PORC shall be responsible for reviewing the effectiveness of corrective actions for:

   1. Licensee Event Reports and Special Reports made to the NRC;

   2. Violations of Technical Specifications;

   3. Special reviews and investigations as requested by the Site Vice President; and

   4. Reports on unplanned onsite releases of radioactive material to the environs.

C. The PORC shall review additional programs, procedures and plant activities as directed by the Site Vice President.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NA	A.1 NA	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.3.1 NA	NA

BASES

The PORC shall be established to recommend to the Station Manager approval or disapproval of the items listed under this commitment prior to their final approval.

The PORC shall report to the Site Vice President on the areas of responsibility specified in this selected licensee commitment.

REFERENCES:

1. ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel
2. ANSI N18.7-1976/ANS-3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
3. Nuclear System Directive 308, Plant Operations Review Committee

16.13 CONDUCT OF OPERATIONS

16.13.4 REACTIVITY ANOMALY

COMMITMENT Report the cause of reactivity anomalies to the Nuclear Regulatory Commission.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.4.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 4.10 during the conversion to ITS.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.5 Additional Operating Shift Requirements

COMMITMENT a. Minimum operating shift staffing shall include the following additional personnel in excess of ITS requirements:

1. One Shift Work Manager when any unit is in MODES 1, 2, 3 or 4
2. One additional Non-Licensed Operator when Unit 1 and 2 are in MODES 1, 2, 3, or 4 and Unit 3 is not in MODES 1, 2, 3, or 4.
3. Two Non-Licensed operator when no units are in MODES 1, 2, 3 or 4.
4. At least one licensed operator shall be in the reactor building when fuel handling operations in the reactor building are in progress.
5. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition those specified in ITS 5.5.2.b and 10 CFR 50.54(m) shall supplement the control room staff.

b. The Shift Work Manager shall be an experienced SRO.

APPLICABILITY: With fuel in any of the three reactor vessels.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.5.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.1.1.9 and Table 6.1-1 during the conversion to ITS.

The requirements of this commitment are in addition to the requiremements specified in ITS 5.2.2. For example, when Units 1 and 2 are in MODES 1, 2, 3, or 4 and Unit 3 is not in MODES 1, 2, 3, or 4 (i.e., two units are operating from one control room), ITS 5.2.2.a requires 4 non-licensed operators. SLC 16.13.5.1.b requires an additional non-licensed operator for a total of five non-licensed operators.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.6 Retraining and Replacement of Station Personnnel

COMMITMENT Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.6.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.1.1.7 during the conversion to ITS.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.7 Procedures for Control of pH in Recirculated Coolant After Loss-of-Coolant Accident (LOCA) & Long-Term Emergency Core Cooling Systems

COMMITMENT

a. Procedures shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours.

b. Procedures shall include provision for remote or local operation of system components necessary to establish high and low pressure injection within 15 minutes after a line break.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.7.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.4.1.i and 6.4.1.k during the conversion to ITS.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.8 Respiratory Protective Program

COMMITMENT A respiratory protective program approved by the Commission shall be in force.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.8.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.4.4.a during the conversion to ITS.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.9 Startup Report

COMMITMENT A summary report of unit startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the facility license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the unit. Startup reports shall be submitted (1) within 90 days following completion of the startup test program, (2) 90 days following resumption of commencement of commercial power operation, or (3) nine months following initial criticality, whichever occurs first. If a startup report does not cover all three events, i.e., initial criticality, completion of the startup test program and resumption or commencement of commercial power operation supplementary reports shall be submitted at least every three months until all three events are completed.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.9.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.6.1.1 during the conversion to ITS.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.10 Core Operating Limits Report

COMMITMENT Concentrated Boric Acid Storage Tank volume and boron concentration limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle and shall be documented in the CORE OPERATING LIMITS REPORTS.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.10.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.9.1.4 during the conversion to ITS.

REFERENCES

N/A

16.13 CONDUCT OF OPERATIONS

16.13.11 Procedure for Station Survey Following an Earthquake

COMMITMENT Written procedures with appropriate check-off lists and instructions shall be provided for a station survey following an earthquake:

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.11.1 N/A	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 6.4.1.f during the conversion to ITS.

REFERENCES

N/A

16.14 CONTROL RODS AND POWER DISTRIBUTION

16.14.1 APSR Movement

COMMITMENT Perform specified SR.

APPLICABILITY: MODES 1 and 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.14.1.1	Verify that loss of power will not cause APSR movement.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of this SLC section were relocated from CTS 4.7.1 during the conversion to ITS.

REFERENCES

N/A

16.14 CONTROL RODS AND POWER DISTRIBUTION

16.14.2 Control Rod Program Verification

COMMITMENT CONTROL RODS shall be operated in their programmed functional position and group.

APPLICABILITY: MODES 1 and 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONTROL ROD not operating in its programmed functional position and group.	A.1 Declare CONTROL ROD inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.14.2.1	Each control rod drive mechanism shall be selected from the control room and exercised by a movement of approximately two inches to verify that the proper rod has responded as shown on the unit computer printout of that rod.	Whenever the control rod drive patch panel is locked (after inspection, test, reprogramming, or maintenance)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.14.2.2	Independently verify that the power or instrumentation cables to control rod drive assemblies atop the reactor are connected.	Upon reconnecting after the cables have been disconnected or removed

BASES

The requirement(s) of this SLC section were relocated from CTS 3.5.2.2.b.6 and CTS 4.7.2 during the conversion to ITS.

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer, identified by a unique number (1 through 69) associated with only one core position. The other set of outputs goes to a programmable bank of 69 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or to the control room meter bank are improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by: (1) selecting a specific rod from one group (e.g., Rod 1 in Regulating Group 6), (2) noting that the program-approved core position for this rod of the group (assume the approved core position is No. 53), (3) exercising of the selected rod and (4) noting that the computer prints out both absolute and relative position response for the approval core position (assumed to be position No. 53) and that the proper meter responds in the control room display bank (assumed to be Rod 1 in Group 6) for both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, it is necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

REFERENCES

1. UFSAR, Section 7.6.

16.14 CONTROL RODS AND POWER DISTRIBUTION

16.14.3 Power Mapping

COMMITMENT Perform specified SRs.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.14.3.1 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution.	-----NOTE------ SR 3.0.2 is not applicable -------------- 10 EFPD

BASES

The requirement(s) of this SLC section were relocated from CTS 4.1.5 during the conversion to ITS.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

REFERENCES

N/A

16.14 CONTROL RODS AND POWER DISTRIBUTION

16.14.4 Control Rod Drive Patch Panels

COMMITMENT — The control rod drive patch panels shall be locked with limited access to be authorized by the manager or his designated alternate.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.14.4.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.5.2.7 during the conversion to ITS.

REFERENCES

N/A

16.15 VENTILATION FILTER TESTING PROGRAM

16.15.1 Penetration Room Ventilation Room System Testing

COMMITMENT Perform specified Surveillance Requirements.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.15.1.1	Verify carbon sample removed from the Penetration Room Ventilation System Filters provide ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, and 95% R.H.).	31 days after removal of carbon sample
SR 16.15.1.2	Verify each Penetration Room Ventilation System fan when tested in accordance with ANSI N510-1975 operates at system design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. -------------- 18 months +25%

(continued)

16.15 VENTILATION FILTER TESTING PROGRAM

16.15.1 Penetration Room Ventilation Room System Testing

COMMITMENT Perform specified Surveillance Requirements.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1 N/A.	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.15.1.1	Verify carbon sample removed from the Penetration Room Ventilation System Filters provide ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, and 95% R.H.).	31 days after removal of carbon sample
SR 16.15.1.2	Verify each Penetration Room Ventilation System fan when tested in accordance with ANSI N510-1975 operates at system design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

(continued)

SURVEILLANCE		FREQUENCY
SR 16.15.1.3	Verify Penetration Room Ventilation System HEPA filters provide ≥ 99% DOP removal when tested in accordance with ANSI N510-1975 at system design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after each complete or partial replacement of a HEPA filter bank AND Once after any structural maintenance on the system housing AND Once after painting, fire, or chemical release in any ventilation zone communicating with the system

(continued)

SURVEILLANCE		FREQUENCY
SR 16.15.1.4	Verify Penetration Room Ventilation System charcoal adsorber filters provide ≥ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975 at system design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after each complete or partial replacement of a charcoal adsorber bank AND Once after any structural maintenance on the system housing AND Once after painting, fire, or chemical release in any ventilation zone communicating with the system

(continued)

SURVEILLANCE		FREQUENCY
SR 16.15.1.5	Remove carbon samples from Penetration Room Ventilation System for laboratory analysis.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after 720 hours of system operation AND Once after painting, fire or chemical release in any ventilation zone communicating with the system
SR 16.15.1.6	Verify pressure drop across the combined HEPA filters and carbon adsorber banks is < 6 inches of water when tested in accordance with ANSI N510-1975 at system design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

BASES

The requirement(s) of a portion of this SLC section were relocated from CTS 4.5.4.1.b.1, 4.5.4.1.c and 4.5.4.1.e during the conversion to ITS. This SLC also includes the Ventilation Filter Testing Program requirements for the Penetration Room Ventilation System specified in ITS 5.5.12, Ventilation Filter Testing Program.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance is as specified, the calculated doses should be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

REFERENCES

1. Regulatory Guide 1.52, Rev. 2.

2. ITS 5.5.12, Ventilation Filter Testing Program.

16.15 VENTILATION FILTER TESTING PROGRAM

16.15.2 Control Room Pressurization and Filtering System

COMMITMENT Perform specified Surveillance Requirements.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A.	A.1.1 N/A.	N/A.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.15.2.1	Verify carbon sample removed from the Control Room Booster Fan train Filters provide ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, and 95% R.H.).	31 days after removal of carbon sample
SR 16.15.2.2	Verify pressure drop across pre-filters is ≤ 1 inch of water and pressure drop across HEPA filters is ≤ 2 inches of water when tested in accordance with ANSI N510-1975 at system design flow (± 10%).	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.15.2.3	Verify Control Room Booster Fan train HEPA filters provide $\geq$ 99.5% DOP removal when tested in accordance with ANSI N510-1975 at system design flow $\pm$ (10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after each complete or partial replacement of a HEPA filter bank AND Once after any structural maintenance on the system housing AND Once after painting, fire, or chemical release in any ventilation zone communicating with the system

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.15.2.4	Verify Control Room Booster Fan train charcoal adsorber filters provide $\geq$ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975 at system design flow ± (10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after each complete or partial replacement of a charcoal adsorber bank AND Once after any structural maintenance on the system housing AND Once after painting, fire, or chemical release in any ventilation zone communicating with the system

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 16.15.2.5	Remove carbon samples from Control Room Booster Fan trains for laboratory analysis	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after 720 hours of system operation AND Once after painting, fire or chemical release in any ventilation zone communicating with the system

BASES

The requirement(s) of this SLC section were relocated from CTS 4.12 during the conversion to ITS. This SLC also includes the Ventilation Filter Testing Program requirements for the Control Room Booster Fan train filters specified in ITS 5.5.12, Ventilation Filter Testing Program.

The purpose of the control room pressurization filtering system is to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine Building or Auxiliary Building only. The system is designed with two 50 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, carbon filters, booster fans, air handling unit fans, and associated ductwork to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available.

Refueling frequency testing of the installed carbon adsorber stage and absolute filters will verify the leak integrity of the cleanup system.

REFERENCES

N/A

16.15 VENTILATION FILTER TESTING PROGRAM

16.15.3 Spent Fuel Pool Ventilation System

COMMITMENT Perform specified Surveillance Requirements

APPLICABILITY: During movement of fuel within the spent fuel storage pool.
During crane operations with loads over the spent fuel storage pools.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. N/A	A.1 N/A	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.15.3.1	Verify carbon sample removed from the Reactor Building Purge Filters provide $\geq$ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, and 95% R.H.).	31 days after removal of carbon sample
SR 16.15.3.2	Verify each Spent Fuel Ventilation fan operates at design flow (± 10%) when tested in accordance with ANSI N510-1975.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25%

(continued)

SURVEILLANCE		FREQUENCY
SR 16.15.3.3	Remove carbon sample from Reactor Building Purge Filters for laboratory analysis.	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% <u>AND</u> Once after 720 hours of system operation <u>AND</u> Once after painting, fire, or chemical release in any ventilation zone communicating with the System

(continued)

SURVEILLANCE		FREQUENCY
SR 16.15.3.4	Verify Reactor Building purge HEPA filters provide $\geq$ 99% DOP removal when tested in accordance with ANSI N510-1975 at design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% <u>AND</u> Once after each complete or partial replacement of HEPA filter bank <u>AND</u> Once after any structural maintenance on the system housing <u>AND</u> Once after painting, fire, or chemical release in any ventilation zone communicating with the system

(continued)

SURVEILLANCE		FREQUENCY
SR 16.15.3.5	Verify Reactor Building purge charcoal adsorber filters provide ≥ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975 at design flow (± 10%).	------NOTE----- The provisions of SLC 16.2.7 do not apply. --------------- 18 months +25% AND Once after each complete or partial replacement of charcoal adsorber bank AND Once after any structural maintenance on the system housing AND Once after painting, fire, or chemical release in any ventilation zone communicating with the system

BASES

The requirement(s) of this SLC section were relocated from CTS 4.14 during the conversion to ITS. This SLC also includes the Ventilation Filter Testing Program requirements for the Spent Fuel Pool Ventilation System specified in ITS 5.5.12, Ventilation Filter Testing Program.

The offsite doses for the fuel handling accident are within the guidelines of 10 CFR 100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

The Unit 2 Reactor Building purge filter is used in the ventilation system for the common spent fuel pool for Units 1 and 2. The Unit 3 Reactor Building purge filter is used in the Unit 3 spent fuel pool ventilation system. Each filter is constructed with a prefilter, an absolute filter and a charcoal filter in series. The high efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine.

Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the doses for a fuel handling accident would be minimized.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the spent fuel pool ventilation system every month will demonstrate operability of the fans, filters and adsorber system.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

<u>REFERENCES</u>

1. Regulatory Guide 1.52, Rev. 2.

2. ITS 5.5.12, Ventilation Filter Testing Program