Duke Energy®

REGIS T. REPKO
Vice President
McGuire Nuclear Station

Duke Energy
MG01VP / 12700 Hagers Ferry Rd.
Huntersville, NC 28078

980-875-4111
980-875-4809 fax
regis.repko@duke-energy.com

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U.S. Nuclear Regulatory Commission
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Subject: Duke Energy Carolinas, LLC
McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370
Selected Licensee Commitment Manual

Attached is a copy of the current McGuire Selected Licensee Commitment (SLC) Manual. The SLC Manual is Chapter 16 of the McGuire Updated Final Safety Analysis Report.

Questions related to this submittal should be directed to Kay Crane, McGuire Regulatory Compliance at (980) 875-4306.

Regis T. Repko

Attachment: McGuire Selected Licensee Commitment (SLC) Manual

A001
NRR

xc w/attachment:

Mr. Jon H. Thompson, Project Manager (Addressee Only)
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

U.S. Nuclear Regulatory Commission Region II
DRP/RPB1
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

xc w/o attachment:

Mr. J.B. Brady
Senior Residént Inspector
U.S. Nuclear Regulatory Commission
McGuire Nuclear Station

Regis T. Repko affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

Regis T. Repko, Site Vice President, McGuire Nuclear Station

Subscribed and sworn to me: 12-13-2010
Date

Kay L. Crane, Notary Public

My commission expires: 4-1-2012
Date

KAY L. CRANE
NOTARY
PUBLIC
LINCOLN COUNTY, N.C.

ATTACHMENT

McGuire Selected Licensee Commitment (SLC) Manual

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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 McGuire Nuclear Station Selected Licensee Commitments Manual. Those previous technical specification requirements which did not meet the criteria are relocated in this chapter. McGuire Technical Specification 5.4 (Procedures and Programs) requires written procedures to be established, implemented, and maintained on these selected licensee commitments.

The control of the McGuire Nuclear Station selected licensee commitment program and manual shall be in accordance with an approved Nuclear System Directive. The manual is officially designated as Chapter 16 of the McGuire UFSAR. The original issue and subsequent revisions of the manual are approved by the station manager. Administrative requirements of the manual are the responsibility of the Regulatory Compliance Section.

Changes to these Selected Licensee Commitments shall be considered a change in an NRC commitment and shall be made only in accordance with the approved Nuclear System Directive for the Control of Selected Licensee Commitments and by use of the 10 CFR 50.59 Process.

Additional operational related 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 testing requirements for certain systems, and remedial actions to be taken in the event the system is not fully capable of performing its design function. 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 and Testing Requirements within UFSAR 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 REMEDIAL ACTION(S) shall be met, except as provided in SLC 16.2.11. If the COMMITMENT is met or is no longer applicable prior to expiration of the specified time interval, completion of the REMEDIAL ACTION(S) is not required, unless otherwise stated.

16.2.3 When a COMMITMENT is not met, except as provided in the associated REMEDIAL ACTIONS, the Station Manager and/or the Responsible Group Superintendent will determine any further actions.

16.2.4 When a COMMITMENT is not met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall not be made except when the associated REMEDIAL ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This COMMITMENT shall not prevent changes in OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with REMEDIAL ACTIONS. Exceptions to this COMMITMENT are stated in the individual COMMITMENTS.

16.2.5 COMMITMENTS including the associated REMEDIAL ACTIONS shall apply to each unit individually unless otherwise indicated as follows:

a. Whenever the COMMITMENT refers to systems or components which are shared by both units, the REMEDIAL ACTIONS will apply to both units simultaneously. This will be indicated in the REMEDIAL ACTIONS;

b. Whenever the COMMITMENT applies to only one unit, this will be identified in the Applicability section of the COMMITMENT; and

c. 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, for example, "...flow rate of 54,000 cfm (Unit 1) or 43,000 cfm (Unit 2)..."

16.2 APPLICABILITY (continued)

16.2.6 Testing Requirements shall be met during the OPERATIONAL MODES or other specified conditions in the Applicability for individual COMMITMENTS unless otherwise stated in an individual Testing Requirement or Reference. Failure to meet a Testing Requirement, whether such failure is experienced during the performance of the Testing Requirement or between performances of the Testing Requirement, shall be failure to meet the COMMITMENT. Failure to perform a Testing Requirement within the specified Frequency shall be failure to meet the COMMITMENT except as provided in COMMITMENT 16.2.8. Exceptions to these requirements are stated in the individual commitments or may be approved by the Station Manager and/or the Responsible Group Superintendent. Testing Requirements do not have to be performed on inoperable equipment or variables outside specified limits.

16.2.7 The specified Frequency for each Testing Requirement is met if the Test is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this SLC are stated in the individual SLCs.

16.2.8 If it is discovered that a Testing Requirement was not performed within its specified Frequency, then compliance with the requirement to declare the COMMITMENT not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Testing Requirement. A risk evaluation shall be performed for any Testing Requirement delayed greater than 24 hours, and the risk impact shall be managed.

If the Testing Requirement is not performed within the delay period, the COMMITMENT must immediately be declared not met, and the applicable REMEDIAL ACTIONS must be entered.

When the Testing Requirement is performed within the delay period and the Testing Requirement is not met, the COMMITMENT must immediately be declared not met, and the applicable REMEDIAL ACTIONS must be entered.

The clarifications provided by the McGuire Nuclear Station Technical Specification Bases for SR 3.0.3 are similarly applicable to SLC 16.2.8 and Testing Requirements.

16.2 APPLICABILITY (continued)

16.2.9 Entry into an OPERATIONAL MODE or other specified condition in the Applicability of a COMMITMENT shall not be made unless the COMMITMENT'S Testing Requirement(s) have been met within the specified frequency or as approved by the Station Manager and/or Responsible Group Superintendent. This provision shall not prevent entry into OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with REMEDIAL ACTIONS.

16.2.10 Testing Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 16.2.5 for individual commitments or whenever certain portions of a specification contain testing parameters different for each unit, which will be identified in parentheses or footnotes.

16.2.11 Equipment removed from service or declared inoperable to comply with REMEDIAL ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to SLC 16.2.2 for the system returned to service under administrative control to perform the required testing to demonstrate OPERABILITY.

16.3 DEFINITIONS

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

AMSAC	ATWS Mitigation System Activation Circuitry, the Westinghouse system for mitigating ATWS events.
ATWS	An ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) is an expected operational transient (such as loss of feedwater, loss of condenser vacuum, or loss of off site power) which is accompanied by a failure of the reactor trip system to shut down the reactor.
COMMITMENT	A COMMITMENT is a method of ensuring the lowest functional capability or performance levels of equipment which are important to the safety of the facility but are not of such a level of importance that they need to be under the rigorous control provided by Technical Specifications.
MEMBER(S) OF THE PUBLIC	MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors of vendors. Also excluded from this category are persons who enter the site to service equipment or make deliveries. This category does not include personnel who use portions of the site for recreational, occupational, or other purposes not associated with the plant.
PROCESS CONTROL PROGRAM (PCP)	The PCP shall contain the correct 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 10CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

16.3 DEFINITIONS (continued)

PURGE or PURGING	PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain, temperature, pressure, humidity, concentration or other operating condition in such a manner that replacement air or gas is required to purify the confinement.
REMEDIAL ACTION	REMEDIAL ACTION shall be that part of a Selected Licensee Commitment which prescribes remedial measures required under designated conditions.
SECURED	Related to valve position indicates that: 1. For manual valves, the subject valve is locked in the desired position, or 2. for automatic valves, the subject valve is de-energized and properly tagged
SITE BOUNDARY	SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.
SOLIDIFICATION	The immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates 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).
SOURCE CHECK	SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity or a simulated source of radioactivity such as a light emitting diode.
UNRESTRICTED AREA	UNRESTRICTED AREA shall be any area or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters, or for industrial, commercial, institutional, and/or recreational purposes.

16.3 DEFINITIONS (continued)

VENTILATION EXHAUST TREATMENT SYSTEM	A VENTILATION EXHAUST TREATMENT SYSTEM shall be 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. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features (ESF) Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.
VENTING	VENTING shall be the controlled process of discharging air or gas from a confinement to maintain, temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.
WASTE GAS HOLDUP SYSTEM	A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.1 Inventory Control-Reduced Inventory Operation

COMMITMENT

A detailed review of each outage schedule that involves operation at reduced inventory shall be conducted looking at evolutions which could perturb the RCS.

AND

The RCS shall be properly vented when steam generator nozzle dams are in use or the RCS cold leg is open $\geq 1 \text{ in}^2$.

AND

The reactor shall be subcritical for at least 7 days, or as specified in Design Study CNDS-0242, or MGDS-0228/CNDS-0218.

---------------------------------------NOTE--
Activities that could perturb the RCS during reduced inventory operation shall require prior notification of the operations shift manager.

APPLICABILITY

RCS level < 60 inches (wide range) with irradiated fuel in the core.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS not properly vented.	A.1 Initiate action to provide the required hot leg vent path.	Immediately
	AND	
	A.2 Suspend all activities that could perturb RCS level or which may reduce the reliability of the operating ND loop.	Immediately

TESTING REQUIREMENTS

None

BASES

Generic Letter 88-17 and NUREG 1410 involve concerns associated with a loss of Residual Heat Removal (RHR) during RCS reduced inventory. Numerous events have occurred in the industry that resulted in a loss of RHR during reduced inventory operation. This is of great concern due to the potential for substantial core damage occurring in a relatively short time period. This SLC depicts those commitments that are extremely important to nuclear safety, however, are not presently covered by Technical Specifications.

Under the stated APPLICABILITY, when EITHER steam generator nozzle dams are in use OR the RCS cold leg side is opened with total opening of one square inch or greater, a hot leg vent path is required. The vent path may be satisfied by:

1. Hot leg nozzle dam <u>not</u> installed on the vented loop

   AND

   Removal of either of the following on the vented loop:

   a. Hot leg diaphragm and manway

      OR

   b. Cold leg diaphragm and manway (with associated cold leg nozzle dam installed*)

OR

2. Reactor vessel head is removed.

*Installation of the associated cold leg nozzle dam avoids diversion of makeup flow (injected via a cold leg nozzle) intended for the core.

REFERENCES

1. Generic Letter 88-17, Loss of Decay Heat Removal

2. NUREG 1410, Loss of Vital AC Power and Residual Heat Removal During Mid-Loop Operation at Vogtle Nuclear Station.

3. Integrated Scheduling Management Procedure 3.1, Outage Planning and Execution Responsibilities

REFERENCES (continued)

4. McGuire Nuclear Station responses to GL 88-17, dated January 3, 1989, February 2, 1989, March 10, 1989 and February 24, 1993.

5. Nuclear Station Directive NSD-403, Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50. 65(a)(4).

6. Design Study CNDS-0242, Catawba and McGuire Nuclear Stations, Shutdown By Decay Heat Level Before Mid-Loop Operation, Safety Analysis, Nuclear Engineering, Nuclear Services.

7. Design Study MGDS-0228/CNDS-0218, McGuire/Catawba Nuclear Stations, Loss of Decay Heat Removal With Steam Generator Mitigation, Safety Analysis, Engineering Support Section, Design Engineering Department.

8. PIP M-08-05725.

9. PIP M-09-04863.

16.5 REACTOR COOLANT SYSTEM

16.5.2 Power Systems and Decay Heat Removal - Reduced Inventory Operation

COMMITMENT

Three power sources and two decay heat removal loops shall be available consisting of :

a. Two ND pumps available with one in operation,

b. Two trains of KC and RN pumps available with flow capacity sufficient to maintain stable core exit temperature, and either

c. Two independent buslines capable of supplying the 4160 V buses via normal or standby 7 KV/4160 V transformers and one D/G capable of supplying a 4160V bus, or

d. One busline capable of supplying one 4160V bus via normal or standby 7KV/4160V transformers and two D/G's and associated 4160V buses.

APPLICABILITY: RCS level < 60 inches (wide range) with irradiated fuel in the core.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Initiate action to restore the necessary power supplies to service.	Immediately
	AND	
	A.2 Suspend all activities that could perturb RCS level or which may reduce the reliability of the operating ND loop.	Immediately
	AND	
	A.3 Enter applicable Conditions and Required Actions of LCO 3.4.8 and LCO 3.9.6 for inoperable ND loops.	Immediately

TESTING REQUIREMENTS

None

BASES

Generic Letter 88-17 and NUREG 1410 involve concerns associated with a loss of Residual Heat Removal during NC system reduced inventory. Numerous events have occurred in the industry that resulted in a loss of residual heat removal during reduced inventory operation. This is of great concern due to the potential for substantial core damage occurring in a relatively short time period. This SLC depicts those commitments that are extremely important to nuclear safety, however, are not presently covered by Technical Specifications.

REFERENCES

1. Generic Letter 88-17, Loss of Decay Heat Removal

2. NUREG 1410, Loss of Vital AC Power and Residual Heat Removal During Mid-Loop Operation at Vogtle Nuclear Station.

3. Integrated Scheduling Management Procedure 3.1, Outage Planning and Execution Responsibilities

4. McGuire Nuclear Station responses to GL 88-17, dated January 3, 1989, February 2, 1989, March 10, 1989 and February 24, 1993.

5. McGuire Station Directive 3.1.3 (MSD403) - Shutdown Risk Management Guidelines.

16.5 REACTOR COOLANT SYSTEM

16.5.3 Reactivity Control – Reduced Inventory Operation

COMMITMENT The following independent sources and makeup paths of borated water must be available:

a. One high head source from NV pump train A or train B taking suction on the FWST and capable of discharging to the RCS, and

b. One low head (gravity) source supplied from the FWST to the RCS.

APPLICABILITY: RCS level < 60 inches (wide range) with irradiated fuel in the core.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Initiate action to restore the required makeup sources.	Immediately
	<u>AND</u>	
	A.2 Suspend all activities that could perturb RCS level or which may reduce the reliability of the operating ND loop.	Immediately

TESTING REQUIREMENTS

None

BASES

Generic Letter 88-17 and NUREG 1410 involve concerns associated with a loss of Residual Heat Removal during NC system reduced inventory. Numerous events have occurred in the industry that resulted in a loss of residual heat removal during reduced inventory operation. This is of great concern due to the potential for substantial core damage occurring in a relatively short time period. This SLC depicts those commitments that are extremely important to nuclear safety, however, are not presently covered by Technical Specifications.

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3. Integrated Scheduling Management Procedure 3.1, Outage Planning and Execution Responsibilities

4. McGuire Nuclear Station responses to GL 88-17, dated January 3, 1989, February 2, 1989, March 10, 1989 and February 24, 1993.

5. McGuire Station Directive 3.1.3 (MSD403) - Shutdown Risk Management Guidelines.

16.5 REACTOR COOLANT SYSTEM

16.5.4 RCS Instrumentation – Reduced Inventory Operation

COMMITMENT

Two independent RCS level indications consisting of any valid combination of Wide Range, Narrow Range, Sightglass, Upper RVLIS (Train A/B), Lower RVLIS (Train A/B) or RCS ultrasonic (Loop A/C) level instrumentation shall be provided. The Wide Range and Narrow Range RCS level instrumentation shall have level alarm setpoints for low and high level with trend capability

<u>AND</u>

The following additional instrumentation shall be provided:

a. Two core exit thermocouples operating while the reactor vessel head is in place, or

b. Two additional independent RCS level indications consisting of any combination of Wide Range, Narrow Range, Sightglass, Lower RVLIS (Train A/B), or RCS ultrasonic (Loop A/C) level instrumentation, when the core exit thermocouples are unavailable.

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

1. If RCS sightglass is being used as one of the alternate RCS level indications, then continuously monitor RCS sightglass level indication and record level at an interval no greater than 15 minutes during normal conditions. Water level monitoring should be capable of being performed either (a) by an operator in the Control Room <u>OR</u> (b) from a location other than the Control Room with provision for providing immediate water level values to an operator in the Control Room if significant changes occur.

2. The OAC computer points for the required thermocouples should be used for trending and alarm.

3. Removal of the last two thermocouples shall occur no sooner than 2 hours prior to reactor vessel head removal. Replacement of at least two thermocouples shall occur within 2 hours after reinstalling the reactor vessel head.

APPLICABILITY: RCS level < 60 inches (wide range) with irradiated fuel in the core.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Initiate action to restore the required instrumentation.	Immediately
	AND	
	A.2 Suspend all activities that could perturb RCS level or change RCS pressure.	Immediately

TESTING REQUIREMENTS

None

BASES

Generic Letter 88-17 and NUREG-1410 involve concerns associated with a loss of Residual heat Removal during NC system reduced inventory. Numerous events have occurred in the industry that resulted in loss of residual heat removal during reduced inventory operation. This is of great concern due to the potential for substantial core damage occurring in a relatively short time period. This SLC depicts those commitments that are extremely important to nuclear safety, however, are not presently covered by Technical Specifications.

REFERENCES

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4. McGuire Nuclear Station responses to GL 88-17, dated January 3, 1989, February 2, 1989, March 10, 1989 and February 24, 1993.

5. McGuire Station Directive 3.1.3 (MSD403) - Shutdown Risk Management Guidelines.

16.5 REACTOR COOLANT SYSTEM

16.5.5 Containment Closure - Reduced Inventory Operation

COMMITMENT The capability to close containment following a loss of RHR shall be assured. Containment closure completion shall be achievable prior to the onset of core boiling in the event ND is lost.

APPLICABILITY: RCS level < 60 inches (wide range) with irradiated fuel in the core.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Initiate action to ensure commitment can be met.	Immediately
	AND	
	A.2 Suspend all activities that could perturb RCS level or which may reduce the reliability of the operating ND loop.	Immediately

TESTING REQUIREMENTS

None

BASES

Generic Letter 88-17 and NUREG 1410 involve concerns associated with a loss of Residual Heat Removal during NC system reduced inventory. Numerous events have occurred in the industry that resulted in a loss of residual heat removal during reduced inventory operation. This is of great concern due to the potential for substantial core damage occurring in a relatively short time period. This SLC depicts those commitments that are extremely important to nuclear safety, however, are not presently covered by Technical Specifications.

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4. McGuire Nuclear Station responses to GL 88-17, dated January 3, 1989, February 2, 1989, March 10, 1989 and February 24, 1993.

5. McGuire Station Directive 3.1.3 (MSD403) - Shutdown Risk Management Guidelines.

16.5 REACTOR COOLANT SYSTEM

16.5.6 Safety Valves - Shutdown

COMMITMENT

One pressurizer code safety valve shall be OPERABLE with lift settings $\geq$ 2435 psig and $\leq$ 2559 psig.

---------------------------------------NOTE---------------------------------------
The lift setting pressure shall correspond to ambient conditions of the valve at normal operating temperature and pressure.

APPLICABILITY

MODE 4 with any RCS cold leg temperature $\leq$ 300°F, and MODE 5.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No pressurizer Code safety valve OPERABLE.	A.1 Suspend all operations involving positive reactivity changes.	Immediately
	AND	
	A.2 Place an OPERABLE RHR loop into operation in the shutdown cooling mode.	Immediately

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.5.6.1	Verify the required pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift setting shall be $\geq$ 2460 psig and $\leq$ 2510 psig.	In accordance with the Inservice Testing Program

BASES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

Demonstration of the safety valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code. This SLC and Technical Specification 3.4.10 allow a +3% and -2% setpoint tolerance is allowed for OPERABILITY. The valves are reset to ±1% during surveillance testing to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Code, Section XI

16.5 REACTOR COOLANT SYSTEM

16.5.7 Chemistry

COMMITMENT The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 16.5.7-1.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more chemistry parameters in excess of its Steady State Limit but within its Transient Limit in MODE 1, 2, 3, or 4.	A.1 Restore the parameter to within steady state limit.	24 hours
B. One or more chemistry parameters in excess of its Transient Limit in MODE 1, 2, 3, or 4. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

(continued)

REMEDIAL ACTIONS (continued)

C. ------------NOTE------------ All Required Actions must be completed whenever this Condition is entered. ------------------------------- RCS chloride or fluoride concentration not within the Steady State Limits for more than 24 hours in any condition other than MODES 1, 2, 3 and 4. <u>OR</u> RCS chloride or fluoride concentration not within Transient Limits in any condition other than MODES 1, 2, 3 and 4.	C.1 Initiate action to reduce the pressurizer pressure to ≤ 500 psig. <u>AND</u> C.2 Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the RCS. <u>AND</u> C.3 Determine that the RCS remains acceptable for continued operation.	Immediately Prior to increasing the pressurizer pressure > 500 psig <u>OR</u> Prior to entry to MODE 4 Prior to increasing the pressurizer pressure > 500 psig <u>OR</u> Prior to entry to MODE 4

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.5.7.1 Verify RCS chemistry is within limits.	72 hours

TABLE 16.5.7-1

RCS CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen [1]	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
Fluoride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

Notes:

1. Oxygen limit and associated Testing Requirement not applicable with $T_{avg} \leq 250$ °F.

BASES

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

The Testing Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take remedial action.

REFERENCES

1. McGuire Nuclear Station UFSAR, Section 18.2.4, Chemistry Control Program.
2. McGuire License Renewal Commitments MCS-1274.00-00-0016, Section 4.6, Chemistry Control Program.

16.5 REACTOR COOLANT SYSTEM

16.5.8 Pressurizer

COMMITMENT The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE----------- All Required Actions must be completed whenever this Condition is entered. ------------------------------- Pressurizer temperature not within limits.	A.1 Restore pressurizer temperature to within limits.	30 minutes
	AND	
	A.2 Perform engineering evaluation to determine effects of the out-of-limit condition on the structural integrity of the pressurizer.	72 hours
	AND	
	A.3 Determine that the pressurizer remains acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Reduce pressurizer pressure to < 500 psig.	36 hours

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.5.8.1	-----NOTE----- Only required to be performed during system heatup or cooldown operations. ----- Verify pressurizer temperatures are within limits.	30 minutes
TR 16.5.8.2	-----NOTE----- Only required to be performed during auxiliary spray operations. ----- Verify spray water temperature differential within limit.	12 hours

BASES

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

> The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200 °F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320 °F, and
>
> System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance within the ASME Code requirements.

16.5 REACTOR COOLANT SYSTEM

16.5.9 Structural Integrity

COMMITMENT The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained.

APPLICABILITY: All MODES.

---NOTE---
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Separate Condition entry is allowed for each component.

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REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Structural integrity of any ASME Code Class 1 component(s) not within limits.	A.1 Restore structural integrity of affected component(s) to within limit.	Prior to increasing RCS temperature > 50°F above minimum temperature required by NDT considerations
	<u>OR</u>	
	A.2 Isolate the affected component(s) from service.	Prior to increasing RCS temperature > 50°F above minimum temperature required by NDT considerations

B. Structural integrity of any ASME Code Class 2 or 3 component(s) not within limits.	B.1 Enter the Operability Determination process to promptly confirm that Structural Integrity is still maintained in the degraded or non-conforming condition.		Immediately
	<u>OR</u>		
	B.2 Declare the affected component(s) inoperable.		Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.5.9.1 Verify the structural integrity of ASME Code Class 1, 2, and 3 components is in accordance with the Inservice Inspection Program.	In accordance with the Inservice Inspection Program

BASES

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

UFSAR Section 3.2, "Classification of Structures, Systems, and Components," defines and correlates code classes, safety classes, and Duke piping classes. In general, Class 1 applies to the reactor coolant pressure boundary. Class 2 applies to safety systems such as emergency core cooling, containment heat removal and cleanup, and containment isolation valves. Class 3 applies to the auxiliary feedwater system, spent fuel cooling, and air cleanup systems like control room ventilation. A complete listing of the Class 1, 2, and 3 systems and components are contained in the UFSAR Chapter 3 Tables.

This SLC applies to one or more ASME Code Class 1, 2, or 3 components in a degraded or nonconforming condition. A degraded or nonconforming condition of a system, structure, or component (SSC) is a condition in which there has been any loss of required quality or functional capability, failure to meet requirements of the regulations, or failure to conform to all aspects of the licensing basis. Degraded or nonconforming conditions of SSCs discovered during the conduct of inservice inspections, maintenance or refueling activities, or during plant operation shall be evaluated to determine the affect on structural integrity. References 4, 5, and 6 contain guidance and evaluation methods to be used in determining structural integrity and operability for ASME Code Class 1, 2, or 3 components. Structural integrity and operability may be restored by repair, replacement, or modification in accordance with ASME Section XI. In some cases, an operability determination may be an acceptable method of confirmation of operability. If structural integrity cannot be established promptly or the results are indeterminate, the component shall be declared inoperable and the appropriate Technical Specification action statement entered.

The NOTE above the REMEDIAL ACTIONS clarifies the application of this SLC. The CONDITIONS of this SLC may be entered independently for each component in a degraded or nonconforming condition.

BASES (continued)

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition and Addenda through Winter 1972.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, 1971 Edition and Addenda through Winter 1972.

2. 10 CFR 50.55a (g), Inservice Inspection Requirements.

3. UFSAR Section 3.2, Classification of Structures, Systems, and Components.

4. NRC RIS 2005-20, Revision 1, Revision to NRC Inspection Manual 9900 Technical Guidance, "Operability Determinations and Functionality Assessments for Resolution of Degraded and Nonconforming Conditions."

5. NRC Generic Letter 90-05, Guidance for Performing Temporary Non-code Repairs of ASME Code Class 1, 2, and 3 Piping.

6. NRC Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.

7. NSD 203, "Operability / Functionality".

16.5 REACTOR COOLANT SYSTEM

16.5.10 Reactor Vessel Head Vent System

COMMITMENT

Two reactor vessel head vent paths, each consisting of two valves in series powered from emergency buses, shall be OPERABLE and closed.

-----------------------------------NOTE-----------------------------------
Reactor head vent system may be aligned to support events where normal and excess letdown are unavailable.

APPLICABILITY

MODES 1, 2, 3, and 4.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor vessel head vent path inoperable.	A.1 Initiate action to close and remove power from all valves in the inoperable flow path.	Immediately
	AND	
	A.2 Restore the inoperable vent path to OPERABLE status.	30 days
B. Two reactor vessel head vent paths inoperable.	B.1 Initiate action to close and remove power from all valves in the inoperable flow paths.	Immediately
	AND	
	B.2 Restore at least one inoperable vent path to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.5.10.1 Cycle each valve in the vent through one complete cycle of full travel from the control room during MODE 5 or 6.	18 months
TR 16.5.10.2 Verify flow through vent paths by venting during MODE 5 or 6.	18 months

BASES

Reactor Vessel Head Vents are provided to exhaust non-condensable gases from the primary system that could inhibit natural circulation core cooling. The Reactor Vessel Head Vent System further functions to provide inventory control for standby shutdown facility events ('A' train only) and for events for which normal and excess letdown are unavailable. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function. (Operability of the pressurizer steam space vent path is provided by ITS 3.4.11 and 3.4.12).

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The Testing Requirement to verify Reactor Vessel Head Vent flowpath is qualitative as no specific size or flow rate is required to exhaust non-condensable gases. The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements”, November 1980.

REFERENCES

1. NUREG-0737, “Clarification of TMI Action Plan Requirements”, Item II.B.1, November 1980.
2. PIP M97-3795.
3. PIP M00-0201.

16.6 ENGINEERED SAFETY FEATURES

16.6.1 Containment Sump

COMMITMENT The containment sump shall be maintained free of loose debris.

APPLICABILITY MODES 1, 2, 3, and 4.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Debris found in containment or in containment sump.	A.1 Remove debris from containment.	Prior to final exit from containment

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.6.1.1 Perform a visual inspection for all accessible areas of the containment and verify no loose debris is present that could be transported to the sump..	Prior to establishing containment integrity
TR 16.6.1.2 Perform a visual inspection for affected areas of the containment and verify no loose debris is present that could be transported to the sump.	At completion of each containment entry after containment integrity is established

BASES

Removal of identified debris from containment or from the containment sump is critical to the function of ECCS systems during the cold leg recirculation phase following a safety injection. Any loose debris (rags, trash, clothing, etc.) left in containment could be transported to the strainers in the containment sump during LOCA conditions causing a loss of suction or restriction of the ECCS pumps.

REFERENCES

None.

16.6 ENGINEERED SAFETY FEATURES

16.6.3 Inlet Door Position Monitoring System

COMMITMENT The Inlet Door Position Monitoring System shall be OPERABLE.

APPLICABILITY MODES 1, 2, 3, and 4.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Inlet Door Position Monitoring System inoperable.	A.1.1 Verify the Ice Bed Temperature Monitoring System is OPERABLE.	Immediately
	<u>AND</u>	
	A.1.2 Verify ice bed temperature $\leq 27°F$.	Once per 4 hours
	<u>AND</u>	
	A.1.3 Restore the Inlet Door Position Monitoring System to OPERABLE status.	14 days
	<u>OR</u>	
	A.2 Restore the Inlet Door Position Monitoring System to OPERABLE status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.6.3.1	Perform a CHANNEL CHECK.	7 days AND Once within 4 hours after receiving a door open alarm
TR 16.6.3.2	Perform a TADOT.	18 months
TR 16.6.3.3	Verify the Inlet Door Position Monitoring System correctly indicates the status of each inlet door.	18 months, when each door is opened and reclosed during testing per TS 3.6.13

BASES

The OPERABILITY of the Inlet Door Position Monitoring System ensures that the capability is available for monitoring the individual inlet door position. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified limits.

TS 3.6.12 requires that the Ice Bed temperature be maintained less than or equal to 27°F. If the Ice Bed temperature chart recorder is out of service, or otherwise incapable of performing its design function, then alternate means, either independently or in combination, can be used to satisfy the 12 hour surveillance of SR 3.6.12.1. The alternate means have proactive components (if any channel is failed or reads greater than 5°F of its associated channel) and includes the following:

1. Obtain manual temperature readings from multiplexer.
2. Use calibrated instrumentation to measure Ice Bed area/basket temperature.
3. Use calibrated thermography gun to measure ice bed area/basket temperature.

REFERENCES

None.

16.6. ENGINEERED SAFETY FEATURES

16.6.4 Safety Injection System Nozzles

COMMITMENT: Reporting of ECCS Injections and Nozzle Usage Factors

APPLICABILITY: MODES 1, 2, and 3.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
The ECCS is actuated and injects water into the Reactor Coolant System and the current value of the usage factor for an affected safety injection nozzle exceeds 0.70.	A Special Report shall be prepared and submitted to the NRC describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided.	90 days

TESTING REQUIREMENTS

None

BASES

Duke letter, "Justification for Continued Operation with Seven Thermal Sleeves Removed," dated December 14, 1983, provided the results of evaluations performed by Westinghouse and Duke as a basis for continued operation of McGuire Units 1 and 2 without thermal sleeves installed in the reactor coolant system nozzles. The NRC's letter and safety evaluation (SER) dated December 30, 1986, concluded that continued operation was acceptable and that the requirements of License Condition (LC) 2.C.(4) had been completed. The NRC staff's acceptance recognized that McGuire's Technical Specification 3/4.5.2 required reporting of the usage factor of each nozzle if the value exceeded 0.70 and the ECCS actuated and injected water into the reactor coolant system. During implementation of Improved Technical Specifications, this reporting requirement was inadvertently removed.

By letter dated June 13, 2000, as supplemented on August 20, 2001 and September 10, 2001, McGuire submitted a proposed License Amendment Request (LAR). This purpose of this LAR was to delete LCs that had previously been completed. In the September 10, 2001 letter, McGuire committed to ensure proper notification is accomplished in the event the usage factor of affected safety injection nozzles exceeds the 0.70 value following an ECCS actuation and injection of water into the reactor coolant system. The NRC issued approved License Amendment 200/181 by letter dated December 5, 2001.

REFERENCES

1, McGuire License Amendment Request dated June 13, 2000, as supplemented by letters August 30, 2001 and September 10, 2001.

2. NRC letter dated December 5, 2001, McGuire Nuclear Station, Units 1 and 2 Issuance of Amendments RE: License Conditions (TAC NOS. MA9297 and MA9298)

16.7 INSTRUMENTATION

16.7.1 ATWS/AMSAC

COMMITMENT The ATWS/AMSAC system shall be OPERABLE.

APPLICABILITY: MODE 1 above 40% RTP.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ATWS/AMSAC system inoperable.	A.1 Restore ATWS/AMSAC system to OPERABLE status. <u>OR</u> A.2 Prepare and submit a Special Report outlining the cause of the malfunction and plans for restoring the system to OPERABLE status.	7 days 37 days

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.7.1.1 Perform a CHANNEL OPERATIONAL TEST on the ATWS/AMSAC system.	18 months.

BASES

None

REFERENCES

1. Final Design Description, ATWS Mitigation System Activation Circuitry, "AMSAC" Original Issue January 23, 1987, as revised.

16.7 INSTRUMENTATION

16.7.2 Seismic Instrumentation

COMMITMENT The seismic monitoring instrumentation shown in Table 16.7.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more seismic monitoring instruments inoperable.	A.1 Restore inoperable instrument to OPERABLE status.	30 days
	<u>OR</u>	
	A.2 Prepare and submit a Special Report outlining the cause of the malfunction and plans for restoring the instrument(s) to OPERABLE status.	40 days
B. Seismic monitoring instruments actuated during a seismic event ≥ 0.01g.	B.1 Restore instrument to OPERABLE status.	Within 24 hours following the seismic event
	<u>AND</u>	
	B.2 Retrieve data from accessible actuated instruments and analyze to determine magnitude of vibratory ground motion.	Within 24 hours following the seismic event
	<u>AND</u>	
		(continued)

REMEDIAL ACTIONS (continued)

B. (continued)	B.3 Prepare and submit a Special Report describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.	10 days

TESTING REQUIREMENTS

-----NOTE-----
Refer to Table 16.7.2-1 to determine which TRs apply for each Seismic Instrumentation.

TEST		FREQUENCY
TR 16.7.2.1	-----NOTE----- CHANNEL CHECK does not include 1IEEVD 1030 or 1IEEVD 1040. ---------- Perform CHANNEL CHECK.	31 days
TR 16.7.2.2	Perform CHANNEL OPERATIONAL TEST.	6 months
TR 16.7.2.3	Perform a CHANNEL CALIBRATION.	18 months

TABLE 16.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS		MEASUREMENT RANGE	REQUIRED CHANNELS	TESTING REQUIREMENTS
1.	Triaxial Accelerographs			
1.a	1IEEVD 1020 (Remote Sensor A) Unit 1 Containment Base Slab	0-2g	1	TR 16.7.2.1 TR 16.7.2.2 TR 16.7.2.3
1.b	1IEEVD 1010 (Remote Sensor B) Unit 1 CA Pump Room, Elev. 716' - 0"	0-2 g	1	TR 16.7.2.1 TR 16.7.2.2 TR 16.7.2.3
1.c	1IEEVD 1000 Control Room	0-2 g	1*	TR 16.7.2.1 TR 16.7.2.2 TR 16.7.2.3
1.d	1IEEVD 1030 Unit 1 Containment Bldg. Elev. 825' - 4", 0^{o}	0-2g	1	TR 16.7.2.2 TR 16.7.2.3
1.e	1IEEVD 1040 Unit 1 Containment Bldg. Elev. 784' - 10", 0^{o}	0-2g	1	TR 16.7.2.2 TR 16.7.2.3

* With control room indication.

BASES

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. The capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 2.

The seismic system records seismic data acquired by three MEMS triaxial accelerograph sensors connected to a network control center located in the control room. Two additional MEMS triaxial accelerograph sensor/recorders are located in Unit 1 containment. These sensor/recorders are stand-alone units and gather additional information during a seismic event. All of the MEMS triaxial sensor/recorders store seismic data in SRAM (Static random access memory) and can download it to a computer for analysis. Subsequent seismic events are likewise automatically captured and made available for data analysis.

The network control center (NCC2002, 1IEECS 1000) in the control room provides on-line monitoring and data retrieval for the three MEMS triaxial accelerometers (sensors) connected to it (1IEEVD 1000, 1010, 1020). The NCC also continuously self-checks all

BASES (continued)

significant functions and power supply status on-line. Any failure will result in an alarm condition on the NCC and a remote alarm sent to the Unit 1 OAC.

The two stand-alone MEMS triaxial accelerograph sensor/recorders located in Unit 1 containment (1IEEVD 1030, 1040) will require the captured seismic data be downloaded to a computer manually, when they become accessible, after a seismic event. They are not required for the channel check every 31 days per TR 16.7.2.1 because they are not needed to detect a seismic event.

REFERENCES

1. Regulatory Guide 1.12, Instrumentation for Earthquakes, Revision 2.

2. 10 CFR Part 100, Appendix A.

16.7 INSTRUMENTATION

16.7.3 Meteorological Instrumentation

COMMITMENT The meteorological monitoring instrumentation channels shown in Table 16.7.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Meteorological monitoring channels inoperable.	A.1 Restore inoperable channel(s) to OPERABLE status.	7 days
	<u>OR</u>	
	A.2 Prepare and submit a Special Report outlining the cause of the malfunction and plans for restoring the channel(s) to OPERABLE status.	17 days

TESTING REQUIREMENTS

-----NOTE-----
Refer to Table 16.7.3-1 to determine which TRs apply for each meteorological monitoring instrumentation.

TEST	FREQUENCY
TR 16.7.3.1 Perform CHANNEL CHECK.	24 hours
TR 16.7.3.2 Perform a CHANNEL CALIBRATION.	6 months.

TABLE 16.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT AND LOCATION	REQUIRED CHANNELS	TESTING REQUIREMENTS
1. Wind Speed		
1.a Meteorological Tower Nominal Elev. 801.88'	1	TR 16.7.3.1 TR 16.7.3.2
1.b Meteorological Tower Nominal Elev. 964.88'	1	TR 16.7.3.1 TR 16.7.3.2
2. Wind Direction		
2.a Meteorological Tower Nominal Elev. 801.88'	1	TR 16.7.3.1 TR 16.7.3.2
2.b Meteorological Tower Nominal Elev. 964.88'	1	TR 16.7.3.1 TR 16.7.3.2
3. Air Temperature - Delta T		
3.a Meteorological Tower Nominal Elev. 798.46' - 962.22'	1	TR 16.7.3.1 TR 16.7.3.2

BASES

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

REFERENCES

1. Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

16.7 INSTRUMENTATION

16.7.4 Loose-Part Detection System

COMMITMENT The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Loose-Part Detection System monitoring channels inoperable.	A.1 Restore inoperable channel(s) to OPERABLE status.	30 days
	<u>OR</u>	
	A.2 Prepare and submit a Special Report outlining the cause of the malfunction and plans for restoring the channel(s) to OPERABLE status.	40 days

TESTING REQUIREMENTS

--NOTE--

The following Testing Requirements are only required for one of the three channels in each monitored area, i.e., reactor lower vessel, reactor upper vessel, and each steam generator.

TEST	FREQUENCY
TR 16.7.4.1 Perform CHANNEL CHECK.	24 hours
TR 16.7.4.2 Perform CHANNEL OPERATIONAL TEST, excluding setpoint verification.	31 days
TR 16.7.4.3 Perform a CHANNEL CALIBRATION.	18 months.

BASES

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the reactor system and avoid or mitigate damage to reactor system components. The allowable out-of-service times and Testing Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

The Testing Requirements on the Loose-Part Detection System are only required on one of the three channels associated with the reactor lower vessel area (channel 1, 2, or 3), one of the three channels associated with the reactor upper vessel area (channel 4, 5, or 6), and one channel associated with each steam generator (channel 8, 9, or 10 for SG-A, channel 12, 13, or 14 for SG-B, channel 16, 17, or 18 for SG-C, and channel 20, 21, or 22 for SG-D) during each required performance.

REFERENCES

1. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

16.7 INSTRUMENTATION

16.7.5 Turbine Overspeed Protection

COMMITMENT At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY MODE 1.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One stop valve or one governor valve per high pressure turbine steam lead Inoperable.	A.1 Restore inoperable valve to OPERABLE status. <u>OR</u> A.2 Close at least one valve in the affected steam lead(s). <u>OR</u> A.3 Isolate the turbine from the steam supply.	72 hours 78 hours 78 hours
B. One reheat stop valve or one reheat intercept valve per LP turbine steam lead inoperable.	B.1 Restore inoperable valve to OPERABLE status. <u>OR</u> B.2 Close at least one valve in the affected steam lead(s). <u>OR</u> B.3 Isolate the turbine from the steam supply.	72 hours 78 hours 78 hours

(continued)

REMEDIAL ACTIONS (continued)

C. Turbine Overspeed Protection System otherwise inoperable.	C.1 Isolate the turbine from the steam supply.	6 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.7.5.1 Perform inservice inspection in accordance with the Turbine Overspeed Reliability Program.	In accordance with the Turbine Overspeed Reliability Program

BASES

This commitment is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive turbine overspeed of the turbine could generate potentially damaging missiles. All Category I structures except the new fuel vault at McGuire, are designed to withstand effects of turbine missiles without any adverse impact on the safety related equipment housed inside (UFSAR 3.5.2.7 and 10.2.3). To assure protection against turbine overspeed a "Turbine Overspeed Reliability Program" is implemented. Tests and inspections associated with this program will be performed in accordance with station procedures, maintenance work requests and/or outage work schedules as appropriate. All deviations from the program or deficiencies identified through the specified maintenance, calibration, or testing activities are evaluated by Duke Power Company to determine if operability of the system has been affected and appropriate action taken such as correcting the deviation or deficiency, performing compensatory action, or removing the turbine from service.

REFERENCES

1. McGuire Nuclear Station UFSAR Section 3.5.2.7 and 10.2.3

16.7 INSTRUMENTATION

16.7.6 Radiation Monitoring for Plant Operations

COMMITMENT The radiation monitoring instrumentation channels shown in Table 16.7.6-1 shall be FUNCTIONAL.

APPLICABILITY As shown in Table 16.7.6-1.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more radiation monitoring channels Alarm/Trip setpoint exceeding value shown in Table 16.7.6-1.	A.1 Adjust setpoint to within the limit. OR A.2 Declare the channel non-functional.	4 hours 4 hours
B. One Containment Atmosphere Gaseous Radioactivity monitoring channel non-functional.	B.1 Verify containment purge system (VP) valves are maintained closed.	Immediately
C. One or more Control Room Air Intake Radioactivity monitoring channel non-functional.	C.1 Monitor at alternate representative location (Unit Vent). OR C.2 If Unit Vent monitor non-functional, obtain grab samples from Unit Vent in accordance with SLC 16.11.7.	Immediately In accordance with SLC 16.11.7.

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more required channels for Spent Fuel Handling Area, Reactor Building Fuel Handling Area or New Fuel Vault Fuel Handling Area Radiation Monitors non-functional.	D.1 Suspend all fuel movement operations in the fuel handling area being monitored until Required Acton D.2 is completed.	Immediately
	AND	
	D.2.1 Provide a portable continuous monitor with same Alarm Setpoint.	Immediately
	OR	
	D.2.2 Provide RP continuous dose rate monitoring.	Immediately
	AND	
	D.3 Restore non-functional monitors to FUNCTIONAL status.	30 days
E. One Spent Fuel Pool Radioactivity monitoring channel non-functional.	E.1 Verify the Fuel Handling Ventilation System (VF) requirements in Technical Specification 3.7.12 are met.	Immediately
F. Condenser Evacuation System Noble Gas Activity Monitor (EMF-33) non-functional.	F.1 Ensure that all N-16 Leakage Monitor (EMF-71, 72, 73, & 74) channels are FUNCTIONAL.	Immediately
G. One or more N-16 Leakage Monitor (EMF-71, 72, 73, & 74) channels non-functional.	G.1 Ensure that the Condenser Evacuation System Noble Gas Activity Monitor (EMF-33) is FUNCTIONAL.	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Condenser Evacuation System Noble Gas Activity Monitor (EMF-33) non-functional. AND One or more N-16 Leakage Monitor (EMF-71, 72, 73, & 74) channels non-functional.	H.1 Initiate action to restore online radiation monitor to FUNCTIONAL.	Immediately
	AND	
	H.2 Perform TS-SR 3.4.13.2.	72 hours

TESTING REQUIREMENTS

--NOTE--

Refer to Table 16.7.6-1 to determine which TRs apply for each Radiation Monitoring channel.

TEST	FREQUENCY
TR 16.7.6.1 Perform CHANNEL CHECK.	12 hours
TR 16.7.6.2 Perform CHANNEL OPERATIONAL TEST.	92 days
TR 16.7.6.3 Perform CHANNEL OPERATIONAL TEST.	184 days
TR 16.7.6.4 Perform a CHANNEL CALIBRATION.	18 months

TABLE 16.7.6-1

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATION

MONITOR	APPLICABLE MODES	REQUIRED CHANNELS	ALARM/TRIP SETPOINT	TESTING REQUIREMENTS
1. Containment Atmosphere Gaseous Radioactivity-High (Low Range EMF-39)	1,2,3,4,5,6	1	Must meet SLC 16.11-6 limits	TR 16.7.6.1 TR 16.7.6.2 TR 16.7.6.4
2. Spent Fuel Pool Radioactivity-High (EMF-42)	With irradiated fuel in fuel storage areas or fuel building	1	≤ 1.7 x 10-4 μCi/ml	TR 16.7.6.1 TR 16.7.6.2 TR 16.7.6.4
3. Spent Fuel Handling Area Radiation Monitor (1EMF-17, 2EMF-4)	With fuel in fuel storage areas or fuel building	1	≤ 15 mR/hr See Note (b)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
4. Reactor Building Fuel Handling Area Radiation Monitor (1EMF-16, 2EMF-3)	6	1	≤ 15 mR/hr See Note (b)	TR 16.7.6.1 TR 16.7.6:3 TR 16.7.6.4
5. New Fuel Vault Fuel Handling Area Radiation Monitors (1EMF-20, 1EMF-21, 2EMF-7, 2EMF-8)	With fuel in New Fuel Vault	1	≤ 15 mR/hr See Note (b)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
6. Control Room Air Intake Radioactivity-High (EMF-43a and 43b)	1,2,3,4,5,6	2 per station.	≤ 3.4 x 10-4 μCi/ml	TR 16.7.6.1 TR 16.7.6.2 TR 16.7.6.4
7. Condenser Evacuation System Noble Gas Activity Monitor (EMF-33)	1	1	See Note (a)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
8. N-16 Leakage Monitor (EMF-71, 72, 73 & 74)	1 (40-100% reactor power)	4 (1/steamline)	See Note (a)	TR 16.7.6.1 TR 15.7.6.3 TR 16.7.6.4

(a) The setpoint is as required by the primary to secondary leak rate monitoring program.
(b) Setpoint can be elevated above 15 mR/hr based upon direction from approved station procedures.

BASES

The FUNCTIONALITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

Remedial Action C.

Control room air intake radioactivity monitoring channels (EMF-43a and 43b) are noble gas beta detection channels. The alternate representative monitoring locations are both unit vent noble gas monitors 1EMF-36L and 2EMF-36L with continuous indication below TRIP 2 ALARM setpoint. During periods of non-functionality of either 1EMF-36L or 2EMF-36L, the alternate monitoring is satisfied by grab sampling of the unit vent in accordance with SLC 16.11.7.

Remedial Action D.

Fuel assemblies are stored and handled in areas of the plant discussed below. Radiation monitoring is provided for these areas to detect excessive radiation levels and will provide an alarm to alert personnel if a potential radiation hazard is present.

1. Unit 1 and 2 Spent Fuel Pool; includes the cask pool area, the new fuel elevator, the fuel transfer tube area and the spent fuel storage are/racks.

2. Unit 1 and 2 Reactor Building; includes the fuel transfer tube area, the reactor core and the refueling canal.

3. Unit 1 and 2 Fuel Building; includes the new fuel vault area.

Performance of Required Acton D.1 shall not preclude completion of movement of a component to a safe position. When a fuel handling area radiation monitor channel becomes non-functional, an alternate means is required for determining dose rate and alerting individuals to excessive radiation levels. This can be accomplished by either a portable monitor with same alarm setpoint located within the area monitored by the inoperable channel or using Radiation Protection personnel performing continuous monitoring of area dose rate using a hand-held dose rate meter. This hand-held meter will not provide an alarm, but relies upon RP personnel to alert individuals of excessive radiation levels.

Certain evolutions may result in a higher gamma dose rate field, resulting in the need to adjust the alarm setpoint above the nominal alarm/trip setpoint (15 mR/hr). An approved station procedure controls adjustment of this setpoint to a higher value that still ensures individuals are alerted to the presence of excessive radiation levels.

Remedial Action F, G and H.

The condenser evacuation system noble gas activity monitor (EMF-33) and main steam line N-16 monitors (EMF- 71, 72, 73, & 74) are used for online monitoring of primary-to-secondary leak rate. These radiation monitors provide the preferred means to accomplish Technical Specification Surveillance SR 3.4.13.2 while in Mode 1. For the condenser evacuation system noble gas activity monitor (EMF-33) or main steam line N-16 monitor to be considered functional for primary to secondary leakage monitoring the monitor must be sensitive to at least 30 gallons per day (GPD) leakage rate.

REFERENCES

1. Technical Specification 3.4.13 - RCS Operational Leakage.
2. NSD-513 - Primary to Secondary Leak Monitoring Program, Revision 5.
3. 10CFR50.68 - Criticality Accident Requirements
4. Duke letter dates July 29, 2004 - RAI Response, TS 3.7.15 and TS 4.3 Changes.
5. NRC Safety Evaluation Report dated March 17, 2005 - Amendments Nos. 225/207
6. MCS-1578-VC-00-0001, DBD for VC/YC System
7. MCTC-1578-VC-R001-001, TAC for Radiation Monitors EMF-43a and EMF-43b

16.7 INSTRUMENTATION

16.7.7 Movable Incore Detectors

COMMITMENT

The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles

APPLICABILITY

When the Movable Incore Detection System is used for:

a. Recalibration of Excore Neutron Flux Detection System,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(z)$ and $F^N\Delta H$.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Movable Incore Detection System inoperable.	A.1 Suspend use of the Movable Incore Detection System for the applicable monitoring or calibration functions.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.7.7.1 Normalize each movable incore detector output.	24 hours

BASES

The OPERABILITY of the movable 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 core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(z)$ or $F^N\Delta H$, a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

The Testing Requirements require that each detector be demonstrated OPERABLE at least once per 24 hours by normalizing detector output when the system is required for the specified activities.

The interval of 24 hours begins when the normalization procedure for the detectors has been initiated, such that each detector is normalized at least once in a given 24 hour period.

REFERENCES

1. WCAP-8648, June 1976.

16.7 INSTRUMENTATION

16.7.8 Explosive Gas Monitoring Instrumentation

COMMITMENT

One hydrogen monitor and two oxygen monitors for the in-service hydrogen recombiner train shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of SLC 16.11.19 are not exceeded.

APPLICABILITY

During WASTE GAS HOLDUP SYSTEM operation.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required hydrogen monitor inoperable.	A.1 Suspend oxygen supply to the recombiner.	Immediately
	AND	
	A.2 Restore the required hydrogen monitor to OPERABLE status.	14 days
B. One required oxygen monitor inoperable.	B.1 Restore the required oxygen monitor to OPERABLE status.	14 days

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two required oxygen monitors inoperable.	C.1.1 Suspend oxygen supply to the recombiner.	Immediately
	AND	
	C.1.2 Isolate all hydrogen inputs to the system.	Immediately
	AND	
	C.1.3 Restore the required oxygen monitor(s) to OPERABLE status.	14 days
	OR	
	C.2.1 Suspend oxygen supply to the recombiner.	Immediately
	AND	
	C.2.2 Suspend Reactor Coolant system degas.	Immediately
	AND	
	C.2.3 Obtain and analyze grab samples of the in service WG Decay Tank for oxygen concentration.	Once per 24 hours
	AND	
	C.2.4 Restore the required oxygen monitor(s) to OPERABLE status.	14 days

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Prepare and submit a Special Report to the NRC explaining why the inoperability was not corrected in the time specified.	30 days

TESTING REQUIREMENTS

	TEST	FREQUENCY
TR 16.7.8.1	Perform CHANNEL CHECK.	12 hours
TR 16.7.8.2	Perform CHANNEL OPERATIONAL TEST.	31 days
TR 16.7.8.3	-----------------------------NOTES---------------------------------- 1. The CHANNEL CALIBRATION of the hydrogen monitor shall include the use of standard gas samples corresponding to alarm setpoints in accordance with the manufacturer's recommendations. 2. The CHANNEL CALIBRATION of the oxygen monitor shall include the use of standard gas samples in accordance with the manufacturer's recommendations. 3. A standard gas sample of nominal 4 volume percent oxygen, balance nitrogen, shall be used in the calibration to check linearity of the oxygen monitor. --- Perform a CHANNEL CALIBRATION.	92 days

BASES

The gas instrumentation is provided for monitoring and controlling the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM.

SLC 16.7.8 requires that one hydrogen and two oxygen monitors per train be OPERABLE in the Waste Gas (WG) System to ensure that explosive gas mixtures are within the limits of SLC 16.11.19 thereby preventing explosive gas concentrations.

Only one recombiner train is in service at a time. Therefore, the requirement for one hydrogen and two oxygen monitors shall apply only to the train in service.

For applicability purposes, WASTE GAS HOLDUP SYSTEM operation is defined as when the system is receiving any hydrogen inputs. Reactor Coolant system (RCS) degas is the recirculation of gases through the Volume Control Tank (VCT) using the Waste Gas Holdup system to remove hydrogen and oxygen from the RCS.

The requirement for oxygen monitors may be satisfied for Train "A" by using two of the following three monitors:

0WGMT5790 0WGMT6210 0WGMT6211

The requirement for hydrogen monitors may be satisfied for Train "A" by using 0WGMT5590.

The requirement for oxygen monitors may be satisfied for Train "B" by using two of the following three monitors:

0WGMT5780 0WGMT6210 0WGMT6211

The requirement for hydrogen monitors may be satisfied for Train "B" by using 0WGMT5580.

Loops 0WGMT5580 and 0WGMT5590 have both hydrogen and oxygen monitoring capability. The oxygen monitoring portion of these two loops shall not be used to satisfy the oxygen monitor requirements of either train because these oxygen monitors measure the oxygen concentration at the recombiner inlet after the addition of bulk oxygen. This is not representative of the Waste Gas System as defined in SLC 16.11.19. These oxygen monitors will be used for the operation of the hydrogen recombiners but will not be used to satisfy the requirements of SLC 16.7.8.

REFERENCES

1. UFSAR, Section 11.3
2. UFSAR, Section 15.7
3. Catalytic Hydrogen Recombiner Operational Manual, MCM-1201.04-0174.

16.7 INSTRUMENTATION

16.7.9 Rod Position Indication System - Shutdown

COMMITMENT

One rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY

MODES 3, 4 and 5 with the reactor trip breakers in the closed position with rods not fully inserted and capable of withdrawal.

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

For testing or trouble shooting, alternate methods may be used to ensure there is no possibility of rod motion. These methods are pulling fuses, sliding links in the rod control cabinets or removal of CRDM head cables. After one of these alternate methods is used, the reactor trip breakers may remain in the closed position.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required rod position indicators inoperable.	Open the reactor trip breakers.	Immediately

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.7.9.1	----------------------NOTE---------------------- Reactor trip breakers shall be closed to perform this test. -- For each rod, verify upon rod withdrawal that one train of the rod position indicating system responds appropriately to rod motion.	Once prior to initial criticality after each removal of reactor vessel head.

BASES

OPERABILITY of the Rod Position Indicating System is defined as its ability to determine rod position within 12 steps when rods are stationary and thereby ensure compliance with the control rod alignment and insertion limits.

TR 16.7.9.1:

Following removal of the reactor vessel head, one train of the Rod Position Indicating System shall be verified to be functioning prior to initial criticality. Verification of one Rod Position Indication train responds appropriately to rod withdrawal provides reasonable assurance that the Rod Position Indication System is accurately indicating rod positions. Rods shall be withdrawn that amount necessary to demonstrate that the position indication system is capable of determining rod position within ± 12 steps. A note is provided to allow the reactor trip breakers to be closed when performing this test.

REFERENCES

PIP M06-1987

16.7 INSTRUMENTATION

16.7.10 Position Indication System – Test Exception

COMMITMENT

The limitations of SLC 16.7-9 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided:

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.

-----------------------------------NOTE-----------------------------------
This requirement is not applicable during initial calibration of the Rod Position Indication System provided: (1) Keff is maintained $\leq$ 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

APPLICABILITY

MODES 3, 4, and 5 during performance of rod drop time measurements.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Position Indication System inoperable. <u>OR</u> More than one bank of rods withdrawn.	A.1 Open reactor trip breakers.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.7.10.1 Verify Demand Position Indication System and the Rod Position Indication Systems agree a. Within 12 steps when the rods are stationary, and b. Within 24 steps during rod motion.	Within 24 hours prior to the start of rod drop time measurements <u>AND</u> Once per 24 hours during rod drop time measurements

BASES

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

REFERENCES

None.

16.7 INSTRUMENTATION

16.7.11 Hydrogen Monitors

COMMITMENT The Hydrogen Monitors shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Hydrogen Monitor channel inoperable.	A.1 Restore channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Prepare and submit a Special Report to the Commission outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the channel to OPERABLE status.	14 days
C. Two Hydrogen Monitor channels inoperable.	C.1 Restore one Hydrogen Monitor channel to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.7.11.1 Perform CHANNEL CALIBRATION.	92 days

BASES

The Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion during accident conditions. With the elimination of the design basis LOCA hydrogen release (Ref. 5), the Hydrogen Monitors are no longer required to mitigate design basis accidents. The Hydrogen Monitors are now classified as Regulatory Guide 1.97, Category 3 instrumentation. The Hydrogen Monitors are used to assess the degree of core damage during a severe accident and confirm that random or deliberate ignition has taken place.

The OPERABILITY of the Hydrogen Monitors ensures that there is sufficient information available on unit parameters to monitor and assess unit status and behavior following an accident. The availability of the Hydrogen Monitors is important so that responses to corrective actions can be observed and the need for, and the magnitude of, further actions can be determined. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit.

These SLC requirements were relocated from the Technical Specifications as a result of License Amendments 227 and 290 for Units 1 and 2, respectively..

REFERENCES

1. Letter from NRC to G.R. PetersonG.R. Peterson, Duke, License Amendments 227and 209 for Units 1 and 2, respectively, dated April 4April 4, 2005.

2. McGuireMcGuire Updated Final Safety Analysis Report Section 1.8.

3. Regulatory Guide 1.97, Rev. 2.

4. NUREG-0737, Supplement 1, “TMI Action Items.”

5. 10 CFR 50.44, “Combustible gas control for nuclear power reactors.”

16.8 ELECTRICAL POWER SYSTEMS

16.8.1 Containment Penetration Conductor Overcurrent Protective Devices

COMMITMENT All containment penetration conductor overcurrent protective devices shown in Table 16.8.1-1 and Table 16.8.1-2 shall be OPERABLE.

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

REMEDIAL ACTIONS

--NOTES--

1. Separate Condition entry is allowed for each penetration circuit.
2. Enter applicable Conditions and Required Actions for systems made inoperable by containment penetration conductor overcurrent devices.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetration overcurrent protection device(s) inoperable.	A.1.1 De-energize the circuit(s) by tripping the associated redundant circuit breaker or removing the redundant fuse(s).	72 hours
	<u>AND</u>	
	A.1.2 Verify the associated redundant protective device(s) to be tripped or removed.	Once per 31 days
	<u>OR</u>	
	A.2.1 De-energize the circuit(s) by racking out the inoperable circuit breaker or removing the inoperable protective device(s).	72 hours
	<u>AND</u>	
	A.2.2 Verify the inoperable device(s) are racked out or removed.	Once per 31 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

TESTING REQUIREMENTS

---NOTE---

1. All containment penetration conductor overcurrent protective devices in Table 16.8.1-1 and Table 16.8.1-2 shall be demonstrated OPERABLE by performance of the following Testing Requirements.

2. TR 16.8.1.1, 16.8.1.2, and 16.8.1.3 are only required to be performed for 10% of the circuit breakers within each voltage level on a rotating basis during each surveillance interval.

TEST		FREQUENCY
TR 16.8.1.1	Perform a CHANNEL CALIBRATION of associated protective relays for medium voltage circuits (4 - 15 kV).	18 months
TR 16.8.1.2	--------------------------NOTE-------------------------- For each circuit breaker found inoperable during functional tests, an additional representative sample of 10% of the defective type shall be functionally tested until no more failures are found, or all of that type have been functionally tested. -- Perform an integrated system functional test on each medium voltage (4 -15 kV) circuit breaker which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.	18 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.8.1.3	-----NOTES----- 1. Circuit breakers selected for functional testing shall be selected on a rotating basis. 2. For each circuit breaker found inoperable during functional tests, an additional representative sample of 10% of all the defective type shall be functionally tested until no more failures are found or all of that type have been functionally tested. 3. Lower voltage circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. ----- Perform a functional test of lower voltage circuit breakers using the nominal trip setpoint and response time values in Table16.8.1-1 and Table 16.8.1-2.	18 months
TR 16.8.1.4	Perform fuse inspection and maintenance program.	18 months
TR 16.8.1.5	Perform inspection and preventive maintenance on each circuit breaker in accordance with manufacturer's recommendations.	60 months

BASES

The tables listed in this commitment were relocated from the McGuire Technical Specifications with the approval of the U.S. Nuclear Regulatory Commission. Any additions, deletions, or revisions to the table are considered a change in a commitment, can only be changed using the 10 CFR 50.59 process, and shall be performed pursuant to applicable procedure.

Containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or by demonstrating the OPERABILITY of Primary and Backup overcurrent protection devices during periodic surveillance. Primary and Backup overcurrent protection devices are redundant to each other.

BASES (continued)

Electrical penetrations serve a mechanical integrity function in forming part of the containment pressure boundary. Redundant protective devices provide a means of maintaining this mechanical integrity, which ensures proper protection assuming a single random failure of one of the protective devices. In the event a Containment Penetration Conductor Overcurrent Protective device becomes inoperable, the affected electrical penetration must be de-energized. The method of de-energization must include the use of at least one protective device that cannot be adversely affected by a single active failure. Acceptable methods of de-energization the circuit(s) are tripping the associated redundant circuit breaker, removing the associated redundant fuses, racking out the inoperable circuit breaker, or removing the inoperable circuit breaker or fuse. Opening the inoperable circuit breaker and verifying all phases are open is not an acceptable means of de-energizing the circuit based on concerns with internal breaker integrity after interrupting a rated fault current.

The 31 day Completion Time to reverify that devices are removed or tripped in inoperable circuits is acceptable considering the fact that the devices are operated under administrative control and the probability of misalignment is low.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Testing of these circuit breakers consists of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time is compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Each manufacturer's molded case and metal clad circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

Fuse testing is in accordance with IEEE Standard 242-1975. This program will detect any significant degradation of the fuses or improperly sized fuses. Safety is further assured by the "fail safe" nature of fuses, that is, if the fuse fails, the circuit will deenergize.

REFERENCES

1. IEEE Standard 242-1975

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1. 6900 VAC-Swgr			
Primary Bkr-RCP1A	5.0	15.4 @ 25A	Reactor Coolant Pump 1A
Backup Bkr-1TA-5	5.0	16.5 @ 20A	
Primary Bkr RCP1B	5.0	15.4 @ 25A	Reactor Coolant Pump 1B
Backup Bkr-1TB-5	5.0	16.5 @ 20A	
Primary Bkr RCP1C	5.0	15.4 @ 25A	Reactor Coolant Pump 1C
Backup Bkr-1TC-5	5.0	16.5 @ 20A	
Primary Bkr RCP1D	5.0	15.4 @ 25A	Reactor Coolant Pump 1D
Backup Bkr-1TD-5	5.0	16.5 @ 20A	
2. 600 VAC-MCC			
1EMXA-2 1D			
Primary Bkr	20	45 or 70 @ 60A *	NC Pump 1C Thermal Barrier Outlet Auto Isol Vlv 1KC345A
Backup Fuse	20	NA	
1EMXA-2 1E			
Primary Bkr	20	45 or 70 @ 60A *	NC Pump 1A Thermal Barrier Outlet Auto Isol Vlv 1KC394A
Backup Fuse	20	NA	
1EMXA-2 2A			
Primary Bkr	20	45 or 70 @ 60A *	Cont Air Return Fan 1A Damper 1RAF-D-2
Backup Fuse	20	NA	
1EMXA-2 2B			
Primary Bkr	20	45 or 70 @ 60A *	N2 to Prt Cont Isol Inside Vlv 1NC54A
Backup Fuse	20	NA	
1EMXA-2 2C			
Primary Bkr	20	45 or 70 @ 60A *	RCP Mtg Brg Oil Fill Isol Vlv 1NC196A
Backup Fuse	20	NA	
1EMXA-2 3A			
Primary Bkr	30	45 or 70 @ 90A *	Accumulator 1A Disch Isol Vlv 1NI54A
Backup Fuse	30	NA	
1EMXA-2 3B			
Primary Bkr	30	45 or 70 @ 90A *	Accumulator 1C Disch Isol Vlv 1NI76A
Backup Fuse	30	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXA-2 3C			
Primary Bkr	20	45 or 70 @ 60A *	Test Hdr Inside Cont Isol Vlv 1NI95A
Backup Fuse	20	NA	
1EMXA-2 4B			
Primary Bkr	20	45 or 70 @ 60A *	PALS Pnl Smple Ret to Cont. Isol Vlv 1WL-1302A
Backup Fuse	20	NA	
1EMXA-2 4C			
Primary Bkr	20	45 or 70 @ 60A *	Accum 1A Vent to 1NC34 for Blkout Vlv 1NI430A
Backup Fuse	20	NA	
1EMXA-2 5A			
Primary Bkr	20	45 or 70 @ 60A *	RN Containment Isolation Vlv 1RN253A
Backup Fuse	20	NA	
1EMXA-2 5B			
Primary Bkr	20	45 or 70 @ 60A *	RN Containment Isolation Vlv 1RN276A
Backup Fuse	20	NA	
1EMXA-2 7A			
Primary Bkr	20	45 or 70 @ 60A *	S/G 1A Upper Shell Sample Cont Isol Vlv 1NM187A
Backup Fuse	20	NA	
1EMXA-2 7B			
Primary Bkr	20	45 or 70 @ 60A *	S/G 1A Blowdown Line Sample Cont Isol Vlv 1NM190A
Backup Fuse	20	NA	
1EMXA-2 7C			
Primary Bkr	20	45 or 70 @ 60A *	SG 1C Upper Shell Sample Cont Isol Vlv 1NM207A
Backup Fuse	20	NA	
1EMXA-2 8A			
Primary Bkr	20	45 or 70 @ 60A *	SG 1C Blowdown Line Line Sample Cont Isol Vlv 1NM210A
Backup Fuse	20	NA	
1EMXA-3 2C			
Primary Bkr	20	45 or 70 @ 60A *	RV Containment Isolation Vlv 1RV76A
Backup Fuse	20	NA	
1EMXA-3 3A			
Primary Bkr	20	45 or 70 @ 60A *	H2 Purge Exhaust Cont Vessel Isol Vlv 1VE5A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXA-3 4A			
Primary Bkr	20	45 or 70 @ 60A *	H2 Skimmer Fan 1A Suction Isol Vlv 1VX1A
Backup Fuse	20	NA	
1EMXA-3 5B			
Primary Bkr	20	45 or 70 @ 60A *	RCDT Pump Disch Cont Isol Vlv 1WL2A
Backup Fuse	20	NA	
1EMXA-3 5C			
Primary Bkr	20	45 or 70 @ 60A *	RCDT Vent Cont Isol Vlv 1WL39A
Backup Fuse	20	NA	
1EMXA-3 6A			
Primary Bkr	20	45 or 70 @ 60A *	RB Sump Pump Disch Cont Isol Vlv 1WL64A
Backup Fuse	20	NA	
1EMXA-3 6B			
Primary Bkr	20	45 or 70 @ 60A *	Cont Vent Unit Condensate Cont Isol Vlv 1WL321A
Backup Fuse	20	NA	
1EMXA-4 1B			
Primary Bkr	20	45 or 70 @ 60A *	NC Pump Seal Return Cont Vlv 1NV94AC
Backup Fuse	20	NA	
1EMXA-4 3C			
Primary Bkr	30	45 or 70 @ 90A *	NC Loop 1C Discharge to ND System Cont Isol Vlv 1ND2A,C
Backup Fuse	30	NA	
1EMXA-5 1B			
Primary Bkr	20	45 or 70 @ 60A *	Pzr Liquid Sample Line Inside Cont Isol Vlv 1NM3A,C
Backup Fuse	20	NA	
1EMXA-5 2C			
Primary Bkr	20	45 or 70 @ 60A *	Pzr Steam Sample Line Inside Cont Isol Vlv 1NM6A,C
Backup Fuse	20	NA	
1EMXA-5 2D			
Primary Bkr	20	45 or 70 @ 60A *	NC Hotleg 1D Sample Line Cont Isol Vlv 1NM25A,C
Backup Fuse	20	NA	
1EMXA-5 3B			
Primary Bkr	20	45 or 70 @ 60A *	NC Hotleg 1A Sample Line Cont Isol Vlv 1NM22A,C
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXB-4 1B			
Primary Bkr	20	45 or 70 @ 60A *	NC Pump 1B Thermal Barrier Outlet Auto Isol Vlv 1KC364B
Backup Fuse	20	NA	
1EMXB-4 1C			
Primary Bkr	20	45 or 70 @ 60A *	NC Pump 1D Thermal Barrier Auto Isol Vlv 1KC413B
Backup Fuse	20	NA	
1EMXB-4 2A			
Primary Bkr	20	45 or 70 @ 60A *	NC Pumps Return Hdr Pend Inside Isol Vlv 1KC424B
Backup Fuse	20	NA	
1EMXB-4 2B			
Primary Bkr	20	45 or 70 @ 60A *	Reactor Bldg Drn Hdr Inside Cont Isol Vlv 1KC429B
Backup Fuse	20	NA	
1EMXB-4 2C			
Primary Bkr	30	45 or 70 @ 90A *	Accumulator 1B Disch Isol Vlv 1NI65B
Backup Fuse	30	NA	
1EMXB-4 3D			
Primary Bkr	30	45 or 70 @ 90A *	Accumulator 1D Disch Isol Vlv 1NI88B
Backup Fuse	30	NA	
1EMXB-4 3E			
Primary Bkr	20	45 or 70 @ 60A *	Hotleg Inj Check 1NI124, 1NI128 Test Isol Vlv 1NI122B
Backup Fuse	20	NA	
1EMXB-4 4A			
Primary Bkr	20	45 or 70 @ 60A *	Cont Air Return Fan 1B Damper 1RAF-D-4
Backup Fuse	20	NA	
1EMXB-4 4C			
Primary Bkr	20	45 or 70 @ 60A *	NI Accum 1A Sample Line Inside Cont Isol Vlv 1NM72B
Backup Fuse	20	NA	
1EMXB-4 5A			
Primary Bkr	20	45 or 70 @ 60A *	NI Accum 1B Sample Line Inside Cont Isol Vlv 1NM75B
Backup Fuse	20	NA	
1EMXB-4 5B			
Primary Bkr	20	45 or 70 @ 60A *	NI Accum 1C Sample Line Inside Cont Isol Vlv 1NM78B
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXB-4 5C			
Primary Bkr	20	45 or 70 @ 60A *	Accum 1B Vent to 1NC32 for Blkout Vlv 1NI431B
Backup Fuse	20	NA	
1EMXB-4 6A			
Primary Bkr	20	45 or 70 @ 60A *	NI Accum 1D Sample Line Inside Cont Isol Vlv 1NM81B
Backup Fuse	20	NA	
1EMXB-4 6B			
Primary Bkr	20	45 or 70 @ 60A *	SG 1B Upper Shell Sample Cont Isol Vlv 1NM197B
Backup Fuse	20	NA	
1EMXB-4 6C			
Primary Bkr	20	45 or 70 @ 60A *	SG 1B Bowdown Line Sample Cont Isol Vlv 1NM200B
Backup Fuse	20	NA	
1EMXB-4 7B			
Primary Bkr	20	45 or 70 @ 60A *	SG 1D Upper Shell Sample Cont Isol Vlv 1NM217B
Backup Fuse	20	NA	
1EMXB-4 7C			
Primary Bkr	20	45 or 70 @ 60A *	SG 1D Blowdown Line Smple Cont Isol Vlv 1NM220B
Backup Fuse	20	NA	
1EMXB-5 1B			
Primary Bkr	20	45 or 70 @ 60A *	RV Containment Isolation Vlv 1RV33B
Backup Fuse	20	NA	
1EMXB-5 1C			
Primary Bkr	20	45 or 70 @ 60A *	H2 Skimmer Fan 1B Suction Isol Vlv 1VX2B
Backup Fuse	20	NA	
1EMXC-1A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1A (Normal Source)
Backup Fuse	200	NA	
1EMXC-2A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1C (Normal Source)
Backup Fuse	200	NA	
1EMXC-3B			
Primary Bkr	30	45 or 70 @ 90A *	Pzr Cavity Booster Fan 1A (Normal Source)
Backup Fuse	30	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXC-3C			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1A (Normal Source)
Backup Fuse	100	NA	
1EMXC-3D			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1C (Normal Source)
Backup Fuse	100	NA	
1EMXC-4C			
Primary Bkr	90	110 or 125 @ 270A*	Containment Air Return Fan No. 1A (CARF-1A)
Backup Fuse	90	NA	
1EMXC-4D			
Primary Bkr	90	110 or 125 @ 270A*	Hydrogen Recombiner No. 1A
Backup Fuse	90	NA	
1EMXC-6A			
Primary Bkr	40	45 or 70 @ 120A*	Containment Pipe Tunnel Booster Fan CPT-BF-1A
Backup Fuse	40	NA	
1EMXC-6B			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Handling Unit 1A
Backup Fuse	30	NA	
1EMXC-6C			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Hdlg Unit 1C
Backup Fuse	30	NA	
1EMXC-6D			
Primary Bkr	90	110 or 125 @ 270A*	Hydrogen Skimmer Fan No. 1A
Backup Fuse	90	NA	
1EMXC-7C			
Primary Bkr	30	45 or 70 @ 90A*	Upper Cont Return Air Fan No. 1C
Backup Fuse	30	NA	
1EMXC-7D			
Primary Bkr	20	45 or 70 @ 60A*	Pzr Pwr Oper Relief Isol Vlv 1NC33A
Backup Fuse	20	NA	
1EMXC-8C			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 1A (Normal Source)
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXC-8D			
Primary Bkr	20	45 or 70 @ 60A*	Upper Containment Return Air Fan No. 1A
Backup Fuse	20	NA	
1EMXD-1A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1B (Normal Source)
Backup Fuse	200	NA	
1EMXD-2A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1D (Normal Source)
Backup Fuse	200	NA	
1EMXD-3B			
Primary Bkr	40	45 or 70 @ 120A*	Containment Pipe Tunnel Booster Fan CPT-BF-1B
Backup Fuse	40	NA	
1EMXD-3C			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1B (Normal Source)
Backup Fuse	100	NA	
1EMXD-3D			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1D (Normal Source)
Backup Fuse	100	NA	
1EMXD-4C			
Primary Bkr	90	110 or 125 @ 270A*	Containment Air Return Fan No. 1B (CARF-1B)
Backup Fuse	90	NA	
1EMXD-4D			
Primary Bkr	90	110 or 125 @ 270A*	Hydrogen Recombiner No. 1B
Backup Fuse	90	NA	
1EMXD-6C			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Hdlg Unit No. 1B
Backup Fuse	30	NA	
1EMXD-6D			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Hdlg Unit No. 1D
Backup Fuse	30	NA	
1EMXD-6E			
Primary Bkr	90	110 or 125 @ 270*	Hydrogen Skimmer Fan No. 1B
Backup Fuse	90	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1EMXD-7B			
Primary Bkr	30	45 or 70 @ 90A*	Upper Cont Return Air Fan No. 1D
Backup Fuse	30	NA	
1EMXD-7C			
Primary Bkr	20	45 or 70 @ 60A*	Pzr No. 1 Pwr Oper Safety Relief Isol Vlv 1NC31B
Backup Fuse	20	NA	
1EMXD-7D			
Primary Bkr	20	45 or 70 @ 60A*	Pzr No. 1 Pwr Oper Safety Relief Isol Vlv 1NC35B
Backup Fuse	20	NA	
1EMXD-8A			
Primary Bkr	30	45 or 70 @ 90A*	PZR Cavity Booster Fan 1B (Normal Source)
Backup Fuse	30	NA	
1EMXD-8B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 1B (Normal Source)
Backup Fuse	20	NA	
1EMXD-8C			
Primary Bkr	20	45 or 70 @ 60A*	Upper Containment Return Air Fan 1B
Backup Fuse	20	NA	
1EMXD-8D			
Primary Bkr	30	45 or 70 @ 90A*	NC Loop 1C Disch to ND System Cont Isol Vlv 1ND1B
Backup Fuse	30	NA	
1MXM-F1A			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 1LR14
Backup Fuse	40	NA	
1MXM-F1B			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 1LR15
Backup Fuse	40	NA	
1MXM-F1C			
Primary Bkr	30	45 or 70 @ 90A*	S/G Compt. Fan 1C
Backup Fuse	30	NA	
1MXM-F1D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A1 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXM-F1E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A2 Blower A
Backup Fuse	20	NA	
1MXM-F2A			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 1LR16
Backup Fuse	40	NA	
1MXM-F2B			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 1LR17
Backup Fuse	40	NA	
1MXM-F2C			
Primary Bkr	25	45 or 70 @ 75A*	Reactor Bldg Equip Hdlg 5 Ton Jib Crane
Backup Fuse	25	NA	
1MXM-F2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A3 Blower A
Backup Fuse	20	NA	
1MXM-F2E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cont AHU 1A4 Blower A
Backup Fuse	20	NA	
1MXM-F3A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cont AHU 1A5 Blower A
Backup Fuse	20	NA	
1MXM-F3B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cont AHU 1A6 Blower A
Backup Fuse	20	NA	
1MXM-F3C			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Room Sump Pump 1
Backup Fuse	20	NA	
1MXM-F3D			
Primary Bkr	100	110 or 150 @ 300A*	Upper Cont Welding Recpt
Backup Fuse	100	NA	
1MXM-F4A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A7 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXM-F4B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A8 Blower A
Backup Fuse	20	NA	
1MXM-F4D			
Primary Bkr	100	110 or 150 @ 300A*	Welding Feeder
Backup Fuse	100	NA	
1MXM-F5C			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Defrost Heater 1A
Backup Fuse	50	NA	
1MXM-F6C			
Primary Bkr	60	110 or 150 @ 180A*	Reactor Coolant Drain Tank Pump 1A
Backup Fuse	60	NA	
1MXM-F7A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A9 Blower A
Backup Fuse	20	NA	
1MXM-F7B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A10 Blower A
Backup Fuse	20	NA	
1MXM-F7C			
Primary Bkr	20	45 or 70 @ 60A*	Lower Cont Aux Charcoal Filter Fan 1A
Backup Fuse	20	NA	
1MXM-F8A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A11 Blower A
Backup Fuse	20	NA	
1MXM-F8B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A12 Blower A
Backup Fuse	20	NA	
1MXM-F8C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A13 Blower A
Backup Fuse	20	NA	
1MXM-R1A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B1 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXM-R1B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B2 Blower A
Backup Fuse	20	NA	
1MXM-R1C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B3 Blower A
Backup Fuse	20	NA	
1MXM-R1D			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1A Oil Lift Pump No. 1
Backup Fuse	30	NA	
1MXM-R2A			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 1LR12
Backup Fuse	40	NA	
1MXM-R2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B4 Blower A
Backup Fuse	20	NA	
1MXM-R2E			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1B Oil Lift Pump No. 1
Backup Fuse	30	NA	
1MXM-R3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B5 Blower A
Backup Fuse	20	NA	
1MXM-R3E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B6 Blower A
Backup Fuse	20	NA	
1MXM-R3F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1C Oil Lift Pump No. 1
Backup Fuse	30	NA	
1MXM-R4D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B7 Blower A
Backup Fuse	20	NA	
1MXM-R4E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B8 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXM-R4F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1D Oil Lift Pump No. 1
Backup Fuse	30	NA	
1MXM-R5B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B9 Blower A
Backup Fuse	20	NA	
1MXM-R5C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B10 Blower A
Backup Fuse	20	NA	
1MXM-R5D			
Primary Bkr	175	200 @ 525A	Ice Cond Equip Pwr Pnlbd 1A
Backup Fuse	175	NA	
1MXM-R6A			
Primary Bkr	20	45 or 70 @ 60A*	Rod Cntrl Cluster Change Fixture Hoist Drive
Backup Fuse	20	NA	
1MXM-R6B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B11 Blower A
Backup Fuse	20	NA	
1MXM-R6E			
Primary Bkr	150	110 or 230 @ 450A*	175 Ton Polar Crane
Backup Fuse	150	NA	
1MXM-R7A			
Primary Bkr	20	45 or 70 @ 60A*	Stud Tensioner Hoist
Backup Fuse	20	NA	
1MXM-R7B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive 1A
Backup Fuse	20	NA	
1MXM-R7D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B12 Blower A
Backup Fuse	20	NA	
1MXM-R7E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B13 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXM-R8A			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive 1B
Backup Fuse	20	NA	
1MXM-R8B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive 1C
Backup Fuse	20	NA	
1MXM-R8D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B14 Blower A
Backup Fuse	20	NA	
1MXM-R8E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B15 Blower A
Backup Fuse	20	NA	
1MXMA-1B			
Primary Bkr	30	45 or 70 @ 90A*	Pzr Cavity Booster Fan 1B (Alt Source)
Backup Fuse	30	NA	
1MXMA-1D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A14 Blower A
Backup Fuse	20	NA	
1MXMA-1E			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 1A Pump 1A1
Backup Fuse	20	NA	
1MXMA-2A			
Primary Bkr	25	45 or 70 @ 75A*	RCPM Maintenance Crane Recpt 1A, 1B, 1C, & 1D
Backup Fuse	25	NA	
1MXMA-2B			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR6
Backup Fuse	25	NA	
1MXMA-2C			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 1LR18
Backup Fuse	40	NA	
1MXMA-2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A15 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXMA-3A			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR9
Backup Fuse	25	NA	
1MXMA-3B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond Equip Access Door 1A
Backup Fuse	20	NA	
1MXMA-3C			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Pump 1A
Backup Fuse	50	NA	
1MXMA-3D			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 1B Pump 1B1
Backup Fuse	20	NA	
1MXN-F1A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A1 Blower B
Backup Fuse	20	NA	
1MXN-F1B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A2 Blower B
Backup Fuse	20	NA	
1MXN-F1C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A3 Blower B
Backup Fuse	20	NA	
1MXN-F1D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A4 Blower B
Backup Fuse	20	NA	
1MXN-F2A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A5 Blower B
Backup Fuse	20	NA	
1MXN-F2B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A6 Blower B
Backup Fuse	20	NA	
1MXN-F2C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A7 Blower B
Backup Fuse	20	NA	
1MXN-F2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A8 Blower B
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXN-F3A			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR1
Backup Fuse	25	NA	
1MXN-F3B			
Primary Bkr	30	45 or 70 @ 90A*	S/G Compt. Fan 1B
Backup Fuse	30	NA	
1MXN-F3C			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR2
Backup Fuse	25	NA	
1MXN-F3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A9 Blower B
Backup Fuse	20	NA	
1MXN-F3E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A10 Blower B
Backup Fuse	20	NA	
1MXN-F4A			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive No 1D
Backup Fuse	20	NA	
1MXN-F4B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive No. 1E
Backup Fuse	20	NA	
1MXN-F4C			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive No. 1F
Backup Fuse	20	NA	
1MXN-F4D			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 1A Pump 1A2
Backup Fuse	20	NA	
1MXN-F5C			
Primary Bkr	60	110 or 150 @ 180A*	Reactor Coolant Drain Tank Pump 1B
Backup Fuse	60	NA	
1MXN-F6B			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 1B Pump 1B2
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXN-F6C			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Defrost Htr 1B
Backup Fuse	50	NA	
1MXN-F7A			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR4
Backup Fuse	25	NA	
1MXN-F7B			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR5
Backup Fuse	25	NA	
1MXN-F7C			
Primary Bkr	20	45 or 70 @ 60A*	Fuel Transfer Sys Reactor Side Fdr
Backup Fuse	20	NA	
1MXN-F7D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A11 Blower B
Backup Fuse	20	NA	
1MXN-F8B			
Primary Bkr	30	45 or 70 @ 90A*	S/G Compt. Fan 1A
Backup Fuse	30	NA	
1MXN-F8D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A12 Blower B
Backup Fuse	20	NA	
1MXN-F8E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A13 Blower B
Backup Fuse	20	NA	
1MXN-R1D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B1 Blower B
Backup Fuse	20	NA	
1MXN-R1E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B2 Blower B
Backup Fuse	20	NA	
1MXN-R1F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1A Oil Lift Pump No. 2
Backup Fuse	30	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXN-R2C			
Primary Bkr	30	45 or 70 @ 90A*	Reactor Cavity Manipulator Crane
Backup Fuse	30	NA	
1MXN-R2F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1B Oil Lift Pump No. 2
Backup Fuse	30	NA	
1MXN-R3A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B3 Blower B
Backup Fuse	20	NA	
1MXN-R3B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B4 Blower B
Backup Fuse	20	NA	
1MXN-R3C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B5 Blower B
Backup Fuse	20	NA	
1MXN-R3D			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1C Oil Lift Pump No. 2
Backup Fuse	30	NA	
1MXN-R4A			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Bridge Crane
Backup Fuse	50	NA	
1MXN-R4B			
Primary Bkr	30	45 or 70 @ 90A*	RB Equip Hatch Hoist No. 1
Backup Fuse	30	NA	
1MXN-R4C			
Primary Bkr	30	45 or 70 @ 90A*	S/G Compt. Fan 1D
Backup Fuse	30	NA	
1MXN-R4D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B6 Blower B
Backup Fuse	20	NA	
1MXN-R4E			
Primary Bkr	30	45 or 70 @ 90A*	RCP 1D Oil Lift Pump No.2
Backup Fuse	30	NA	
1MXN-R5D			
Primary Bkr	175	200 @ 525A	Ice Cond Equip Pwr Pnlbd 1B

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	175	NA	
1MXN-R6A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B7 Blower B
Backup Fuse	20	NA	
1MXN-R6B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B8 Blower B
Backup Fuse	20	NA	
1MXN-R6C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B9 Blower B
Backup Fuse	20	NA	
1MXN-R6D			
Primary Bkr	100	110 or 150 @ 300A*	Welding Fdr
Backup Fuse	100	NA	
1MXN-R7A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B10 Blower B
Backup Fuse	20	NA	
1MXN-R7B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B11 Blower B
Backup Fuse	20	NA	
1MXN-R7C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B12 Blower B
Backup Fuse	20	NA	
1MXN-R7D			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Pump 1B
Backup Fuse	50	NA	
1MXN-R8D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B13 Blower B
Backup Fuse	20	NA	
1MXN-R8E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B14 Blower B
Backup Fuse	20	NA	
1MXN-R8F			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1B15 Blower B
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1MXNA-2A			
Primary Bkr	30	45 or 70 @ 90A*	Pzr Cavity Booster Fan 1A (Alt Source)
Backup Fuse	30	NA	
1MXNA-2B			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR7
Backup Fuse	25	NA	
1MXNA-2C			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 1LR8
Backup Fuse	25	NA	
1MXNA-2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A14 Blower B
Backup Fuse	20	NA	
1MXNA-2E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 1A15 Blower B
Backup Fuse	20	NA	
1MXNA-3A			
Primary Bkr	20	45 or 70 @ 60A*	2 Ton CRDM Hdlg Jib Crane
Backup Fuse	20	NA	
1MXNA-3C			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump Motor Drain Tank Pump No. 1
Backup Fuse	20	NA	
1MXNA-3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond Equip Access Door B
Backup Fuse	20	NA	
SMXA-F4A			
Primary Bkr	15	45 or 70 @ 45A*	Unit 1 Emergency Personnel Lock
Backup Fuse	15	NA	
SMXC-7D			
Primary Bkr	15	45 or 70 @ 45A*	Unit 1 Personnel Lock
Backup Fuse	15	NA	
SMXG-F3G			
Primary Bkr	20	45 or 70 @ 60A*	Standby Makeup Pump to Cont Sump Isol Vlv 1NV1012C
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
SMXG-F4G			
Primary Bkr	20	45 or 70 @ 60A*	Standby Makeup Pump to NC Pump Seals Isol Vlv 1NV1013C
Backup Fuse	20	NA	
SMXG-F5A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 28, 55, & 56
Backup Fuse	90	NA	
3. 600 VAC-Press Htr Pwr Pnls			
Backup Press Htr Pwr Pnl 1A-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 1, 2, & 22
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1A-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 5, 6, & 27
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1A-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 9, 10, & 32
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1A-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 11, 12, & 35
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1A-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 13, 14, & 37
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1A-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 17, 18, & 42
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1B-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 21, 47, & 48
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1B-1B			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 26, 53, & 54
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1B-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 31, 59, & 60
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1B-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 36, 65, & 66
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1B-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 41, 71, & 72
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1B-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 46, 77, & 78
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1C-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 7, 8, & 30
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1C-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 19, 20, & 45
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1C-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 24, 51, & 52
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1C-1D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 29, 57, & 58
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1C-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 34, 63, & 64
Backup Fuse	90	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Press Htr Pwr Pnl 1C-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 39, 69, & 70
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1C-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 44, 75, & 76
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1D-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 3, 4, & 25
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1D-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 15, 16, & 40
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1D-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 23, 49, & 50
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1D-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 33, 61, & 62
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1D-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 38, 67, & 68
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 1D-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 43, 73, & 74
Backup Fuse	90	NA	
4. 120 VAC-Panelboards			
1KM-1			
Primary Bkr	30	45 or 70 @ 90*	RCP 1A Space Htr
Backup Fuse	30	NA	
1KM-2			
Primary Bkr	30	45 or 70 @ 90*	RCP 1C Space Htr

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	30	NA	
1KN-1			
Primary Bkr	30	45 or 70 @ 90*	RCP 1B Space Htr
Backup Fuse	30	NA	
1KN-2			
Primary Bkr	30	45 or 70 @ 90*	RCP 1D Space Htr
Backup Fuse	30	NA	
1KN-27			
Primary Bkr	20	36 or 70 @ 60*	Fuel Handling Control Console
Backup Fuse	20	NA	
1KN-31			
Primary Bkr	20	36 or 70 @ 60*	Incore Inst. 120 VAC Outlet Receptacles
Backup Fuse	20	NA	
5. 250 VDC-Lighting			
RB Deadlight Pnlbd 1DLD # 1			
Primary Bkr	20	40 @ 60	Ltg Pnl Nos. 1LR1 & 1LR2
Backup Fuse	20	NA	
RB Deadlight Pnlbd 1DLD # 3			
Primary Bkr	20	40 @ 60	Ltg Pnl Nos. 1LR4, 1LR5, & 1LR6
Backup Fuse	20	NA	
RB Deadlight Pnlbd 1DLD # 4			
Primary Bkr	20	40 @ 60	Ltg Pnl Nos. 1LR7, 1LR8, & 1LR9
Backup Fuse	20	NA	
RB Deadlight Pnlbd 1DLD # 6			
Primary Bkr	20	40 @ 60	Ltg Pnl Nos. 1LR12
Backup Fuse	20	NA	
RB Deadlight Pnlbd 1DLD # 7			
Primary Bkr	20	40 @ 60	Ltg Pnl Nos. 1LR16
Backup Fuse	20	NA	
RB Deadlight Pnlbd 1DLD # 9			
Primary Bkr	20	40 @ 60	Ltg Pnl Nos. 1LR18 & 1LR17
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
6. VAC - LC			
Outage Power Fuse Box 1A			
Primary Fuse	150	NA	Upper Containment Safety Switch 1A
Backup Fuse	150		
Primary Fuse	150	NA	Upper Containment Safety Switch 1B
Backup Fuse	150	NA	
Primary Fuse	200	NA	Lower Containment Outage Power Distr. Pnl 1A
Backup Fuse	200	NA	
Primary Fuse	225	NA	Lower Containment Outage Power Distr. Pnl 1B
Backup Fuse	225	NA	
7. 600 VAC - Containment HVAC Alternate Feeders			
1VTB-1A			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 1A (Alt Source)
1VTF-1A			
Primary Fuse	20	NA	Incore Instrumentation Rm Air Hdlg Unit 1A (Alt Source)
1VTB-1B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 1B (Alt Source)
1VTF-1B			
Primary Fuse	20	NA	Incore Instrumentation Rm Air Hdlg Unit 1B (Alt Source)
1VRB-1A			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1A (Alt Source)
1VRF-1A			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 1A (Alt Source)
1VRB-1B			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1B (Alt Source)

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1VRF-1B			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 1B (Alt Source)
1VRB-1C			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 1C (Alt Source)
1VRF-1C			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 1C (Alt Source)
1VRB-1D			
Primary Bkr	100	110 or 150@ 300A*	Control Rod Drive Vent Fan No. 1D (Alt Source)
1VRF-1D			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 1D (Alt Source)
1VLB-1A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1A (Alt Source)
1VLF-1A			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 1A (Alt Source)
1VLB-1B			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1B (Alt Source)
1VLF-1B			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 1B (Alt Source)
1VLB-1C			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 1C (Alt Source)
1VLF-1C			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 1C (Alt Source)
1VLB-1D			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-1
UNIT 1 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
			Unit No. 1D (Alt Source)
1VLF-1D			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 1D (Alt Source)

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
1. 6900 VAC-Swgr			
Primary Bkr-RCP2A	5.0	15.4 @ 25A	Reactor Coolant Pump 2A
Backup Bkr-2TA-5	5.0	16.5 @ 20A	
Primary Bkr RCP2B	5.0	15.4 @ 25A	Reactor Coolant Pump 2B
Backup Bkr-2TB-5	5.0	16.5 @ 20A	
Primary Bkr RCP2C	5.0	15.4 @ 25A	Reactor Coolant Pump 2C
Backup Bkr-2TC-5	5.0	16.5 @ 20A	
Primary Bkr RCP2D	5.0	15.4 @ 25A	Reactor Coolant Pump 2D
Backup Bkr-2TD-5	5.0	16.5 @ 20A	
2. 600 VAC-MCC			
2EMXA-2 1D			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump 2C Thermal Barrier Outlet Auto Isol Vlv 2KC345A
Backup Fuse	20	NA	
2EMXA-2 1E			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump 2A Thermal Barrier Outlet Auto Isol Vlv 2KC394A
Backup Fuse	20	NA	
2EMXA-2 2A			
Primary Bkr	20	45 or 70 @ 60A*	Cont Air Return Fan 2A Damper 2RAF-D-2
Backup Fuse	20	NA	
2EMXA-2 2B			
Primary Bkr	20	45 or 70 @ 60A*	N2 to Prt Cont Isol Inside Vlv 2NC54A
Backup Fuse	20	NA	
2EMXA-2 2C			
Primary Bkr	20	45 or 70 @ 60A*	RCP Mtg Brg Oil Fill Isol Vlv 2NC196A
Backup Fuse	20	NA	
2EMXA-2 3A			
Primary Bkr	30	45 or 70 @ 90A*	Accumulator 2A Disch Isol Vlv 2N154A
Backup Fuse	30	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2EMXA-2 3B			
Primary Bkr	30	45 or 70@ 90A*	Accumulator 2C Disch Isol Vlv 2NI76A
Backup Fuse	30	NA	
2EMXA-2 3C			
Primary Bkr	20	45 or 70 @ 60A*	Test Hdr Inside Cont Isol Vlv 2NI95A
Backup Fuse	20	NA	
2EMXA-2 4B			
Primary Bkr	20	45 or 70 @ 60A*	PALS Pnl Smple Ret to Cont. Isol Vlv 2WL-1302A
Backup Fuse	20	NA	
2EMXA-2 4C			
Primary Bkr	20	45 or 70 @ 60A*	Accum 2A Vent to 2NC34 for Blkout Vlv 2NI430A
Backup Fuse	20	NA	
2EMXA-2 5A			
Primary Bkr	20	45 or 70 @ 60A*	RN Containment Isolation Vlv 2RN253A
Backup Fuse	20	NA	
2EMXA-2 5B			
Primary Bkr	20	45 or 70 @ 60A*	RN Containment Isolation Vlv 2RN276A
Backup Fuse	20	NA	
2EMXA-2 7A			
Primary Bkr	20	45 or 70 @ 60A*	S/G 2A Upper Shell Sample Cont Isol Vlv 2NM187A
Backup Fuse	20	NA	
2EMXA-2 7B			
Primary Bkr	20	45 or 70 @ 60A*	S/G 2A Blowdown Line Sample Cont Isol Vlv 2NM190A
Backup Fuse	20	NA	
2EMXA-2 7C			
Primary Bkr	20	45 or 70 @ 60A*	SG 2C Upper Shell Sample Cont Isol Vlv 2NM207A
Backup Fuse	20	NA	
2EMXA-2 8A			
Primary Bkr	20	45 or 70 @ 60A*	SG 2C Blowdown Line Line

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	20	NA	Sample Cont Isol Vlv 2NM210A
2EMXA-3 1A			
Primary Bkr	20	45 or 70 @ 60A*	Lower Cont Vent Unit discharge cont isol vlv 2RV76A
Backup Fuse	20	NA	
2EMXA-3 3A			
Primary Bkr	20	45 or 70 @ 60A*	H2 Purge Exhaust Cont Vessel Isol Vlv 2VE5A
Backup Fuse	20	NA	
2EMXA-3 4A			
Primary Bkr	20	45 or 70 @ 60A*	H2 Skimmer Fan 2A Suction Isol Vlv 2VX1A
Backup Fuse	20	NA	
2EMXA-3 5B			
Primary Bkr	20	45 or 70 @ 60A*	RCDT Vent Cont Isol Vlv 2WL2A
Backup Fuse	20	NA	
2EMXA-3 5C			
Primary Bkr	20	45 or 70 @ 60A*	RCDT Vent Cont Isol Vlv 2WL39A
Backup Fuse	20	NA	
2EMXA-3 6A			
Primary Bkr	20	45 or 70 @ 60A*	RB Sump Pump Disch Cont Isol Vlv 2WL64A
Backup Fuse	20	NA	
2EMXA-3 6B			
Primary Bkr	20	45 or 70 @ 60A*	Cont Vent Unit Condensate Cont Isol Vlv 2WL321A
Backup Fuse	20	NA	
2EMXA-4 1B			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump Seal Return Cont Vlv 2NV94AC
Backup Fuse	20	NA	
2EMXA-4 3C			
Primary Bkr	30	45 or 70 @ 90A*	NC Loop 2C Discharge to ND System Cont Isol Vlv 2ND2A,C
Backup Fuse	30	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2EMXA-5 1B			
Primary Bkr	20	45 or 70 @ 60A*	Pzr Liquid Sample Line Inside Cont Isol Vlv 2NM3A,C
Backup Fuse	20	NA	
2EMXA-5-2C			
Primary Bkr	20	45 or 70 @ 60A*	Pzr Steam Sample Line Inside Cont Isol Vlv 2NM6A,C
Backup Fuse	20	NA	
2EMXA-5 2D			
Primary Bkr	20	45 or 70 @ 60A*	NC Hotleg 2D Sample Line Cont Isol Vlv 2NM25A,C
Backup Fuse	20	NA	
2EMXA-5 3B			
Primary Bkr	20	45 or 70 @ 60A*	NC Hotleg 2A Sample Line Cont Isol Vlv 2NM22A,C
Backup Fuse	20	NA	
2EMXB-4 1B			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump 2B Thermal Barrier Outlet Auto Isol Vlv 2KC364B
Backup Fuse	20	NA	
2EMXB-4 1C			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump 2D Thermal Barrier Auto Isol Vlv 2KC413B
Backup Fuse	20	NA	
2EMXB-4 2A			
Primary Bkr	20	45 or 70 @ 60A*	NC Pumps Return Hdr Pend Inside Isol Vlv 2KC424B
Backup Fuse	20	NA	
2EMXB-4 2B			
Primary Bkr	20	45 or 70 @ 60A*	Reactor Bldg Drn Hdr Inside Cont Isol Vlv 2KC429B
Backup Fuse	20	NA	
2EMXB-4 2C			
Primary Bkr	30	45 or 70 @ 90A*	Accumulator 2B Disch Isol Vlv 2NI65B
Backup Fuse	30	NA	
2EMXB-4 3D			
Primary Bkr	30	45 or 70 @ 90A*	Accumulator 2D Disch Isol Vlv

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	30	NA	2NI88B
2EMXB-4 3E			
Primary Bkr	20	45 or 70 @ 60A*	Hotleg Inj Check 2NI124, 2NI128 Test Isol Vlv 2NI122B
Backup Fuse	20	NA	
2EMXB-4 4A			
Primary Bkr	20	45 or 70 @ 60A*	Cont Air Return Fan 2B Damper 2RAF-D-4
Backup Fuse	20	NA	
2EMXB-4 4C			
Primary Bkr	20	45 or 70 @ 60A*	NI Accum 2A Sample Line Inside Cont Isol Vlv 2NM72B
Backup Fuse	20	NA	
2EMXB-4 5A			
Primary Bkr	20	45 or 70 @ 60A*	NI Accum 2B Sample Line Inside Cont Isol Vlv 2NM75B
Backup Fuse	20	NA	
2EMXB-4 5B			
Primary Bkr	20	45 or 70 @ 60A*	NI Accum 2C Sample Line Inside Cont Isol Vlv 2NM78B
Backup Fuse	20	NA	
2EMXB-4 5C			
Primary Bkr	20	45 or 70 @ 60A*	Accum 2B Vent to 2NC32 for Blkout Vlv 2NI431B
Backup Fuse	20	NA	
2EMXB-4 6A			
Primary Bkr	20	45 or 70 @ 60A*	NI Accum 2D Sample Line Inside Cont Isol Vlv 2NM81B
Backup Fuse	20	NA	
2EMXB-4 6B			
Primary Bkr	20	45 or 70 @ 60A*	SG 2B Upper Shell Sample Cont Isol Vlv 2NM197B
Backup Fuse	20	NA	
2EMXB-4 6C			
Primary Bkr	20	45 or 70 @ 60A*	SG 2B Bowdown Line Sample Cont Isol Vlv 2NM200B
Backup Fuse	20	NA	
2EMXB-4 7B			
Primary Bkr	20	45 or 70 @ 60A*	SG 2D Upper Shell Sample Cont

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	20	NA	Isol Vlv 2NM217B
2EMXB-4 7C			
Primary Bkr	20	45 or 70 @ 60A*	SG 2D Blowdown Line Sample Cont Isol Vlv 2NM220B
Backup Fuse	20	NA	
2EMXB-5 1B			
Primary Bkr	20	45 or 70 @ 60A*	Lower cont vent unit supply cont isol vlv 2RV33B
Backup Fuse	20	NA	
2EMXB-5 1C			
Primary Bkr	20	45 or 70 @ 60A*	H2 Skimmer Fan 2B Suction Isol Vlv 2VX2B
Backup Fuse	20	NA	
2EMXC-1A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2A (Normal Source)
Backup Fuse	200	NA	
2EMXC-2A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2C (Normal Source)
Backup Fuse	200	NA	
2EMXC-3C			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2A (Normal Source)
Backup Fuse	100	NA	
2EMXC-3D			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2C (Normal Source)
Backup Fuse	100	NA	
2EMXC-4C			
Primary Bkr	90	110 or 125 @ 270A*	Containment Air Return Fan No. 2A (CARF-2A)
Backup Fuse	90	NA	
2EMXC-4D			
Primary Bkr	90	110 or 125 @ 270A*	Hydrogen Recombiner No. 2A
Backup Fuse	90	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2EMXC-6A			
Primary Bkr	40	45 or 70 @ 120A*	Containment Pipe Tunnel Booster Fan CPT-BF-2A
Backup Fuse	40	NA	
2EMXC-6B			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Handling Unit 2A
Backup Fuse	30	NA	
2EMXC-6C			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Hdlg Unit 2C
Backup Fuse	30	NA	
2EMXC-6D			
Primary Bkr	90	110 or 125 @ 270A*	Hydrogen Skimmer Fan No. 2A
Backup Fuse	90	NA	
2EMXC-7A			
Primary Bkr	30	45 or 70 @ 90A*	PZR Cavity Booster Fan 2A (Normal Source)
Backup Fuse	30	NA	
2EMXC-7B			
Primary Bkr	20	45 or 70 @ 60A*	Upper Containment Return Air Fan No. 2A
Backup Fuse	20	NA	
2EMXC-7C			
Primary Bkr	20	45 or 70 @ 60A*	Upper Cont Return Air Fan No. 2C
Backup Fuse	20	NA	
2EMXC-7D			
Primary Bkr	20	45 or 70 @ 60A*	Pzr Pwr Oper Relief Isol Vlv 2NC33A
Backup Fuse	20	NA	
2EMXC-8C			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 2A (Normal Source)
Backup Fuse	20	NA	
2EMXD-1A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	200	NA	No. 2B (Normal Source)
2EMXD-2A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2D (Normal Source)
Backup Fuse	200	NA	
2EMXD-3B			
Primary Bkr	40	45 or 70 @ 120A*	Containment Pipe Tunnel Booster Fan CPT-BF-2B
Backup Fuse	40	NA	
2EMXD-3C			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2B (Normal Source)
Backup Fuse	100	NA	
2EMXD-3D			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2D (Normal Source)
Backup Fuse	100	NA	
2EMXD-4C			
Primary Bkr	90	110 or 125 @ 270A*	Containment Air Return Fan No. 2B (CARF-2B)
Backup Fuse	90	NA	
2EMXD-4D			
Primary Bkr	90	110 or 125 @ 270A*	Hydrogen Recombiner No. 2B
Backup Fuse	90	NA	
2EMXD-5C			
Primary Bkr	20	45 or 70 @ 60A*	Upper Containment Return Air Fan 2B
Backup Fuse	20	NA	
2EMXD-6C			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Hdlg Unit No. 2B
Backup Fuse	30	NA	
2EMXD-6D			
Primary Bkr	30	45 or 70 @ 90A*	Upper Containment Air Hdlg Unit No. 2D
Backup Fuse	30	NA	
2EMXD-6E			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Bkr	100	110 or 150 @ 300A*	Hydrogen Skimmer Fan No. 2B
Backup Fuse	100	NA	
2EMXD-7B			
Primary Bkr	20	45 or 70 @ 60A*	Upper Cont Return Air Fan No. 2D
Backup Fuse	20	NA	
2EMXD-7C			
Primary Bkr	20	45 or 70 @ 60A*	Pzr Pwr Oper Safety Relief Isol Vlv 2NC31B
Backup Fuse	20	NA	
2EMXD-7D			
Primary Bkr	20	45 or 70 @ 60A*	Pzr Pwr Oper Safety Relief Isol Vlv 2NC35B
Backup Fuse	20	NA	
2EMXD-8A			
Primary Bkr	40	45 or 70 @ 120A*	PZR Cavity Booster Fan 2B (Normal Source)
Backup Fuse	40	NA	
2EMXD-8B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 2B (Normal Source)
Backup Fuse	20	NA	
2EMXD-8D			
Primary Bkr	30	45 or 70 @ 90A*	NC Loop 2C Disch to ND System Cont Isol Vlv 2ND1B
Backup Fuse	30	NA	
2MXM-F1C			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Defrost Heater 2A
Backup Fuse	50	NA	
2MXM-F2A			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR14
Backup Fuse	40	NA	
2MXM-F2B			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR15
Backup Fuse	40	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2MXM-F2C			
Primary Bkr	30	45 or 70 @ 90A*	Pzr Cavity Booster Fan 2B (Alt Source)
Backup Fuse	30	NA	
2MXM-F2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A1 Blower A
Backup Fuse	20	NA	
2MXM-F2E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A2 Blower A
Backup Fuse	20	NA	
2MXM-F3A			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR16
Backup Fuse	40	NA	
2MXM-F3B			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR17
Backup Fuse	40	NA	
2MXM-F3C			
Primary Bkr	25	45 or 70 @ 75A*	Reactor Bldg Equip Hdlg 5 Ton Jib Crane
Backup Fuse	25	NA	
2MXM-F3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A3 Blower A
Backup Fuse	20	NA	
2MXM-F3E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cont AHU 2A4 Blower A
Backup Fuse	20	NA	
2MXM-F4A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cont AHU 2A5 Blower A
Backup Fuse	20	NA	
2MXM-F4B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cont AHU 2A6 Blower A
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2MXM-F4C			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Room Sump Pump
Backup Fuse	20	NA	
2MXM-F4D			
Primary Bkr	100	110 or 150 @ 300A*	Upper Cont Welding Recpt
Backup Fuse	100	NA	
2MXM-F5A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A7 Blower A
Backup Fuse	20	NA	
2MXM-F5B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A8 Blower A
Backup Fuse	20	NA	
2MXM-F5D			
Primary Bkr	100	110 or 150 @ 300A*	Welding Feeder
Backup Fuse	100	NA	
2MXM-F6A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A9 Blower A
Backup Fuse	20	NA	
2MXM-F6B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A10 Blower A
Backup Fuse	20	NA	
2MXM-F6C			
Primary Bkr	30	45 or 70 @ 90A*	Lower Cont Aux Charcoal Filter Fan 2A
Backup Fuse	30	NA	
2MXM-F7A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A11 Blower A
Backup Fuse	20	NA	
2MXM-F7B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A12 Blower A

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	20	NA	
2MXM-F7C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A13Blower A
Backup Fuse	20	NA	
2MXM-F8C			
Primary Bkr	60	110 or 150 @ 180A*	Reactor Coolant Drain Tank Pump 2A
Backup Fuse	60	NA	
2MXM-R1A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B1 Blower A
Backup Fuse	20	NA	
2MXM-R1B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B2 Blower A
Backup Fuse	20	NA	
2MXM-R1C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B3 Blower A
Backup Fuse	20	NA	
2MXM-R1D			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2A Oil Lift Pump No. 1
Backup Fuse	30	NA	
2MXM-R2A			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR12
Backup Fuse	40	NA	
2MXM-R2C			
Primary Bkr	20	45 or 70 @ 160A*	RCPM Maintenance Crane Recpt 2A, 2B, 2C, & 2D
Backup Fuse	20	NA	
2MXM-R2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B4 Blower A
Backup Fuse	20	NA	
2MXM-R2E			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2B Oil Lift Pump No. 1

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	30	NA	
2MXM-R3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B5 Blower A
Backup Fuse	20	NA	
2MXM-R3E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B6 Blower A
Backup Fuse	20	NA	
2MXM-R3F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2C Oil Lift Pump No. 1
Backup Fuse	30	NA	
2MXM-R4D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B7 Blower A
Backup Fuse	20	NA	
2MXM-R4E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B8 Blower A
Backup Fuse	20	NA	
2MXM-R4F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2D Oil Lift Pump No. 1
Backup Fuse	30	NA	
2MXM-R5B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B9 Blower A
Backup Fuse	20	NA	
2MXM-R5C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B10 Blower A
Backup Fuse	20	NA	
2MXM-R5D			
Primary Bkr	175	200 @ 525A	Ice Cond Equip Pwr Pnlbd 2A
Backup Fuse	175	NA	
2MXM-R6A			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Bkr	20	45 or 70 @ 60A*	Rod Cntrl Cluster Change Fixture Hoist Drive
Backup Fuse	20	NA	
2MXM-R6B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B11 Blower A
Backup Fuse	20	NA	
2MXM-R6D			
Primary Bkr	150	110 or 230 @ 450A*	175 Ton Polar Crane
Backup Fuse	150	NA	
2MXM-R7A			
Primary Bkr	20	45 or 70 @ 60A*	Stud Tensioner Hoist
Backup Fuse	20	NA	
2MXM-R7B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive 2A
Backup Fuse	20	NA	
2MXM-R7C			
Primary Bkr	30	45 or 70 @ 90A*	S/G Comp 2D Fan
Backup Fuse	30	NA	
2MXM-R7D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B12 Blower A
Backup Fuse	20	NA	
2MXM-R7E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B13 Blower A
Backup Fuse	20	NA	
2MXM-R8A			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive 2B
Backup Fuse	20	NA	
2MXM-R8B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive 2C
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2MXM-R8C			
Primary Bkr	30	45 or 70 @ 90A*	S/G Comp 2A Fan
Backup Fuse	30	NA	
2MXM-R8D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B14 Blower A
Backup Fuse	20	NA	
2MXM-R8E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B15 Blower A
Backup Fuse	20	NA	
2MXMA-1D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A14 Blower A
Backup Fuse	20	NA	
2MXMA-1E			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 2A Pump 2A1
Backup Fuse	20	NA	
2MXMA-2B			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR6
Backup Fuse	40	NA	
2MXMA-2C			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2R18
Backup Fuse	40	NA	
2MXMA-2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A15 Blower A
Backup Fuse	20	NA	
2MXMA-3A			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 2LR9
Backup Fuse	25	NA	
2MXMA-3B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond Equip Access Door 2A

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	20	NA	
2MXMA-3C			
Primary Bkr	50	110 or 150 @ 150*	Ice Cond Floor Cooling Pump 2A
Backup Fuse	50	NA	
2MXMA-3D			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 2B Pump 2B1
Backup Fuse	20	NA	
2MXN-F1C			
Primary Bkr	60	110 or 150 @ 180A*	Reactor Coolant Drain Tank Pump 2B
Backup Fuse	60	NA	
2MXN-F2A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A1 Blower B
Backup Fuse	20	NA	
2MXN-F2B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A2 Blower B
Backup Fuse	20	NA	
2MXN-F2C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A3 Blower B
Backup Fuse	20	NA	
2MXN-F2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A4 Blower B
Backup Fuse	20	NA	
2MXN-F3A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A5 Blower B
Backup Fuse	20	NA	
2MXN-F3B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A6 Blower B
Backup Fuse	20	NA	
2MXN-F3C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A7 Blower B

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	20	NA	
2MXN-F3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A8 Blower B
Backup Fuse	20	NA	
2MXN-F4A			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 2LR1
Backup Fuse	25	NA	
2MXN-F4B			
Primary Bkr	30	45 or 70 @ 90A*	S/G Comp 2C Fan
Backup Fuse	30	NA	
2MXN-F4C			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 2LR2
Backup Fuse	25	NA	
2MXN-F4D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A9 Blower B
Backup Fuse	20	NA	
2MXN-F4E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A10 Blower B
Backup Fuse	20	NA	
2MXN-F5A			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive No 2D
Backup Fuse	20	NA	
2MXN-F5B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive No. 2E
Backup Fuse	20	NA	
2MXN-F5C			
Primary Bkr	20	45 or 70 @ 60A*	Incore Inst Drive No. 2F
Backup Fuse	20	NA	
2MXN-F5D			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 2A Pump 2A2
Backup Fuse	20	NA	
2MXN-F6A			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 2LR4
Backup Fuse	25	NA	
2MXN-F6B			
Primary Bkr	40	45 or 70 @ 120A*	Lighting Pnlbd 2LR5
Backup Fuse	40	NA	
2MXN-F6C			
Primary Bkr	20	45 or 70 @ 60A*	Fuel Transfer Sys Reactor Side Fdr
Backup Fuse	20	NA	
2MXN-F6D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A11 Blower B
Backup Fuse	20	NA	
2MXN-F7B			
Primary Bkr	30	45 or 70 @ 90A*	Pzr Cavity Booster Fan 2A (Alt Source)
Backup Fuse	30	NA	
2MXN-F7D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A12 Blower B
Backup Fuse	20	NA	
2MXN-F7E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A13 Blower B
Backup Fuse	20	NA	
2MXN-F8B			
Primary Bkr	20	45 or 70 @ 60A*	Cont Floor & Equip Sump 2B Pump 2B2
Backup Fuse	20	NA	
2MXN-F8C			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Defrost Htr 2B
Backup Fuse	50	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2MXN-R1D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B1 Blower B
Backup Fuse	20	NA	
2MXN-R1E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B2 Blower B
Backup Fuse	20	NA	
2MXN-R1F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2A Oil Lift Pump No. 2
Backup Fuse	30	NA	
2MXN-R2C			
Primary Bkr	30	45 or 70 @ 90A*	Reactor Cavity Manipulator Crane
Backup Fuse	30	NA	
2MXN-R2F			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2B Oil Lift Pump No. 2
Backup Fuse	30	NA	
2MXN-R3A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B3 Blower B
Backup Fuse	20	NA	
2MXN-R3B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B4 Blower B
Backup Fuse	20	NA	
2MXN-R3C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B5 Blower B
Backup Fuse	20	NA	
2MXN-R3D			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2C Oil Lift Pump No. 2
Backup Fuse	30	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2MXN-R4A			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Bridge Crane
Backup Fuse	50	NA	
2MXN-R4B			
Primary Bkr	30	45 or 70 @ 90A*	RB Equip Hatch Hoist
Backup Fuse	30	NA	
2MXN-R4C			
Primary Bkr	25	45 or 70 @ 75A*	S/G Comp 2B Fan
Backup Fuse	25	NA	
2MXN-R4D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B6 Blower B
Backup Fuse	20	NA	
2MXN-R4E			
Primary Bkr	30	45 or 70 @ 90A*	RCP 2D Oil Lift Pump No.2
Backup Fuse	30	NA	
2MXN-R5D			
Primary Bkr	175	200 @ 525A	Ice Cond Equip Pwr Pnlbd 2B
Backup Fuse	175	NA	
2MXN-R6A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B7 Blower B
Backup Fuse	20	NA	
2MXN-R6B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B8 Blower B
Backup Fuse	20	NA	
2MXN-R6C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B9 Blower B
Backup Fuse	20	NA	
2MXN-R6D			
Primary Bkr	100	110 or 150 @ 300A*	Welding Fdr
Backup Fuse	100	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2MXN-R7A			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B10 Blower B
Backup Fuse	20	NA	
2MXN-R7B			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B11 Blower B
Backup Fuse	20	NA	
2MXN-R7C			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B12 Blower B
Backup Fuse	20	NA	
2MXN-R7D			
Primary Bkr	50	110 or 150 @ 150A*	Ice Cond Floor Cooling Pump 2B
Backup Fuse	50	NA	
2MXN-R8D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B13 Blower B
Backup Fuse	20	NA	
2MXN-R8E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B14 Blower B
Backup Fuse	20	NA	
2MXN-R8F			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2B15 Blower B
Backup Fuse	20	NA	
Backup Fuse	30	NA	
2MXNA-2B			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 2LR7
Backup Fuse	25	NA	
2MXNA-2C			
Primary Bkr	25	45 or 70 @ 75A*	Lighting Pnlbd 2LR8
Backup Fuse	25	NA	
2MXNA-2D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A14 Blower B

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Fuse	20	NA	
2MXNA-2E			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond AHU 2A15 Blower B
Backup Fuse	20	NA	
2MXNA-3A			
Primary Bkr	20	45 or 70 @ 60A*	2 Ton CRDM Hdlg Jib Crane
Backup Fuse	20	NA	
2MXNA-3C			
Primary Bkr	20	45 or 70 @ 60A*	NC Pump Motor Drain Tank Pump No. 2
Backup Fuse	20	NA	
2MXNA-3D			
Primary Bkr	20	45 or 70 @ 60A*	Ice Cond Equip Access Door 2B
Backup Fuse	20	NA	
SMXD-3E			
Primary Bkr	15	45 or 70 @ 45*	Unit 2 Personnel Lock
Backup Fuse	15	NA	
SMXG-R3G			
Primary Bkr	20	45 or 70 @ 60A*	Standby Makeup Pump to Cont Sump Isol Vlv 2NV1012C
Backup Fuse	20	NA	
SMXG-R4F			
Primary Bkr	20	45 or 70 @ 60A*	Standby Makeup Pump to NC Pump Seals Isol Vlv 2NV1013C
Backup Fuse	20	NA	
SMXV-2F			
Primary Bkr	15	45 or 70 @ 45A*	Unit 2 Emergency Personnel Lock
Backup Fuse	15	NA	
3. 600 VAC-Press Htr Pwr Pnls			
Backup Press Htr Pwr Pnl 2A-1A			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 1, 2, & 22
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2A-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 5, 6, & 27
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2A-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 9, 10, & 32
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2A-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 11, 12, & 35
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2A-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 13, 14, & 37
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2A-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 17, 18, & 42
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2B-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 21, 47, & 48
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2B-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 26, 53, & 54
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2B-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 31, 59, & 60
Backup Fuse	90	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Press Htr Pwr Pnl 2B-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 36, 65, & 66
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2B-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 41, 71, & 72
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2B-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 46, 77, & 78
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2C-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 7, 8, & 30
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2C-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 19, 20, & 45
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2C-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 24, 51, & 52
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2C-1D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 29, 57,.& 58
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2C-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 34, 63, & 64
Backup Fuse	90	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Press Htr Pwr Pnl 2C-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 39, 69, & 70
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2C-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 44, 75, & 76
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2D-1A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 3, 4, & 25
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2D-1B			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 15, 16, & 40
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2D-1C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 23, 49, & 50
Backup Fuse	90	NA	
SMXG-R5A			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 28, 55, & 56
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2D-2C			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 33, 61, & 62
Backup Fuse	90	NA	
Backup Press Htr Pwr Pnl 2D-2D			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 38, 67, & 68
Backup Fuse	90	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Backup Press Htr Pwr Pnl 2D-2E			
Primary Bkr	90	110 or 125 @ 270A*	Pressurizer Heaters 43, 73, & 74
Backup Fuse	90	NA	
4. 120 VAC-Panelboards			
2KM-19			
Primary Bkr	20	45 or 70 @ 60A*	RCP 2A Space Htr
Backup Fuse	20	NA	
2KM-20			
Primary Bkr	20	45 or 70 @ 60A*	RCP 2C Space Htr
Backup Fuse	20	NA	
2KN-19			
Primary Bkr	20	45 or 70 @ 60A*	RCP 2B Space Htr
Backup Fuse	20	NA	
2KN-20			
Primary Bkr	20	45 or 70 @ 60A*	RCP 2D Space Htr
Backup Fuse	20	NA	
2KN-27			
Primary Bkr	20	36 or 70 @ 60A*	Fuel Handling Control Console
Backup Fuse	20	NA	
5. 250 VDC-Lighting			
RB Deadlight Pnlbd 2DLD # 1			
Primary Bkr	20	40 @ 60A	Ltg Pnl Nos. 2LR1 & 2LR2
Backup Fuse	20	NA	
RB Deadlight Pnlbd 2DLD # 3			
Primary Bkr	20	40 @ 60A	Ltg Pnl Nos. 2LR4, 2LR5, & 2LR6
Backup Fuse	20	NA	

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
RB Deadlight Pnlbd 2DLD # 4			
Primary Bkr	20	40 @ 60A	Ltg Pnl Nos. 2LR7, 2LR8, & 2LR9
Backup Fuse	20	NA	
RB Deadlight Pnlbd 2DLD # 6			
Primary Bkr	20	40 @ 60A	Ltg Pnl Nos. 2LR12
Backup Fuse	20	NA	
RB Deadlight Pnlbd 2DLD # 7			
Primary Bkr	20	40 @ 60A	Ltg Pnl Nos. 2LR16
Backup Fuse	20	NA	
RB Deadlight Pnlbd 2DLD # 9			
Primary Bkr	20	40 @ 60A	Ltg Pnl Nos. 2LR18 & 2LR17
Backup Fuse	20	NA	
6. VAC - LC			
Outage Power Fuse Box 2A			
Primary Fuse	150	NA	Upper Containment Safety Switch 2A
Backup Fuse	150	NA	
Ltg Pnl Nos. 2LR1 & 2LR2			
Primary Fuse	150	NA	Upper Containment Safety Switch 2B
Backup Fuse	150	NA	
Primary Fuse	200	NA	Lower Containment Outage Power Distr Pnl 2A
Backup Fuse	200	NA	
Primary Fuse	225	NA	Lower Containment Outage Power Distr Pnl 2B
Backup Fuse	225	NA	
7. 600 VAC - Containment HVAC Alternate Feeders			
2VTB-2A			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 2A (Alt Source)
2VTF-2A			
Primary Fuse	20	NA	Incore Instrumentation Rm Air Hdlg Unit 2A (Alt Source)
2VTB-2B			
Primary Bkr	20	45 or 70 @ 60A*	Incore Instrumentation Rm Air Hdlg Unit 2B (Alt Source)
2VTF-2B			
Primary Fuse	20	NA	Incore Instrumentation Rm Air Hdlg Unit 2B (Alt Source)
2VRB-2A			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2A (Alt Source)
2VRF-2A			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 2A (Alt Source)
2VRB-2B			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2B (Alt Source)
2VRF-2B			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 2B (Alt Source)
2VRB-2C			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2C (Alt Source)
2VRF-2C			
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 2C (Alt Source)
2VRB-2D			
Primary Bkr	100	110 or 150 @ 300A*	Control Rod Drive Vent Fan No. 2D (Alt Source)
2VRF-2D			

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

TABLE 16.8.1-2
UNIT 2 Containment Penetration Conductor Overcurrent Protective Devices

DEVICE NO. & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
Primary Fuse	100	NA	Control Rod Drive Vent Fan No. 2D (Alt Source)
2VLB-2A			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2A (Alt Source)
2VLF-2A			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 2A (Alt Source)
2VLB-2B			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2B (Alt Source)
2VLF-2B			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 2B (Alt Source)
2VLB-2C			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2C (Alt Source)
2VLF-2C			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 2C (Alt Source)
2VLB-2D			
Primary Bkr	200	250 @ 600A	Lower Containment Cooling Unit No. 2D (Alt Source)
2VLF-2D			
Primary Fuse	200	NA	Lower Containment Cooling Unit No. 2D (Alt Source)

* HFB or HFD Circuit Breaker Test Response Time, Respectively.

16.8 ELECTRICAL POWER SYSTEMS

16.8.2 Switchyard Activities

COMMITMENT: Switchyard activities that may affect the availability and reliability of offsite power shall be identified as important to safe plant operation.

APPLICABILITY: Modes 1 through 6

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Abnormal equipment operating conditions and/or alignment.	A.1 Restore equipment to normal conditions and/or alignments.	As soon as possible

TESTING REQUIREMENTS

None

BASES

From the probabilistic risk assessment of McGuire Units 1 and 2, it may be concluded that it is important to minimize the risk of a loss of offsite power (LOOP) event, and it is important to be able to restore offsite power following a LOOP event. The identified risk significant activities are a result of an engineering review to determine those systems or actions that are significant to help maximize the availability and reliability of offsite power. The activities are combinations or alignment and design considerations and good practices, as well as lessons learned from past industry events that have been initiators of or contributors to LOOP events.

This SLC was created to provide a method of tracking the switchyard systems for the purposes of supporting WPM 607 (Maintenance Rule Assessment of Equipment Out of Service) and 10 CFR 50.65 (Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants).

BASES (continued)

Switchyard activities that may affect the availability and reliability of off site power include the following:

230KV Switchyard

1. Work on equipment within the Unit 1 busline boundary. The busline boundary includes the structures, supporting structures, bus, equipment, and hardware from the high side windings of the Unit 1 main step-up transformers up to and including the busline breakers 8, 9, 11 and 12 and their associated disconnects.

2. Work on the protective relaying and/or controls and their cables for equipment within the Unit 1 busline boundary.

525KV Switchyard

1. Work on equipment within the Unit 2 busline boundary. The busline boundary includes the structures, supporting structures, bus, equipment, and hardware from the high side windings of the Unit 2 main step-up transformers up to and including the busline breakers 58, 59, 61 and 62 and their associated disconnects.

2. Work on the protective relaying and/or controls and their cables for equipment within the Unit 2 busline boundary.

Shared Systems between 230KV and 525KV Switchyards

1. Work on the following Switchyard AC auxiliary equipment: (Reference One Line Diagram MC 801-02)

   a. AC load centers and their associated transformers:
      MCOESILXSTA
      MCOESILXSTB
      MCOESILXSTC
      MCOESILXSTD

   b. AC load center feeder circuits to Unit 1 and/or Unit 2 busline boundary equipment:
      Panelboard SPA breakers 6 and 7
      Panelboard SPB breakers 8 and 9
      Panelboard SPC breakers 7, 8 and 9
      Panelboard SPD breaker 11

   c. AC load center feeder circuits to battery chargers:
      Panelboard SPA breaker 23
      Panelboard SPB breaker 23
      Panelboard SPC breaker 18

BASES (continued)

2. Work on the following Switchyard 125V DC system equipment that includes the batteries, chargers, distribution bus, and panelboards. This equipment excludes the panelboard feeders which are addressed in #3 below: (Reference MC 802-01).

   a. Batteries:
      MC0ESHBASY1
      MC0ESHBASY2

   b. Battery Chargers:
      MC0ESHBCSY1
      MC0ESHBCSY2
      MC0ESHBCSY3

   c. 125 VDC Switchyard Distribution Centers:
      SY-DC1
      SY-DC2

   d. 125 VDC Switchyard Panelboards:

DYA	DYB	DYC	DYD	DYE	DYF
DYG	DYH	DYI	DYJ	DYK	DYL
DYM	DYN	DYO	DYP		

3. Work on the Switchyard 125V DC panelboard feeders that serve the Unit 1 and/or Unit 2 busline boundary equipment: (Reference MC 802-01)

   a. 125 VDC Switchyard Panelboard Feeder Breakers:
      DYA - 11, 12
      DYB - 5, 6
      DYC - 1, 2, 3, 4, 16, 17, 18, 19
      DYD - 1, 3, 4, 5, 6
      DYE - 9, 10, 12, 13
      DYF - 4, 5, 6, 7, 14, 15, 16, 17
      DYH - 1, 3, 4, 5, 6
      DYI - 11, 12
      DYJ - 5, 6, 13, 14, 15, 18, 19
      DYK - 20
      DYL - 5, 6, 7, 8, 13
      DYM - 9, 10, 12, 13, 18, 19, 20
      DYN - 14, 15, 16
      DYO - 20
      DYP - 6, 7, 9, 10

<u>General</u>

Cranes or other heavy equipment that have the potential to touch or affect any or all of the four buslines as they are moved in or out of the switchyard, or anywhere within the switchyard where they could touch or affect the buslines.

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 8

2. McGuire Nuclear Station, Technical Specifications and Bases Section 3.8

3. Nuclear System Directive 409, Nuclear Generation Department/Power Delivery Department Switchyard Interface Agreement

4. Nuclear System Directive 502, Corporate Conduct of Operations in the Switchyard

5. McGuire Units 1 and 2 PRA Risk Significant SSC's for the Maintenance Rule, MCC 1535.00-00-0006, SAAG File 208

6. NSAC-203 (EPRI), Losses of Off-Site Power at U. S. Nuclear Power Plants through 1993.

7. MC-801-02 One Line Diagram 230/525KV Switchyard 480/277 AC Load Centers (Rev 21)

8. MC-802-01 One Line Diagram 230/525KV Switchyard 125V DC System (Rev 27)

9. WPM 607, Maintenance Rule Assessment of Equipment Out of Service

10. 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.

16.8 ELECTRICAL POWER SYSTEMS

16.8.3 Diesel Generator (DG) Supplemental Testing Requirements

COMMITMENT: The DG supplemental testing requirements specified below shall be met.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6

-----NOTE-----
The testing requirements for the DG batteries are not required in MODES 5 and 6.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Declare DG inoperable.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.8.3.1 Verify the electrolyte level of each DG battery is above the plates.	7 days
TR 16.8.3.2 Verify overall DG battery voltage is ≥ 125 volts under a float charge.	7 days
TR 16.8.3.3 Verify DG batteries and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.8.3.4	Verify DG battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.	18 months
TR 16.8.3.5	Perform DG battery service test	18 months
TR 16.8.3.6	Remove accumulated water from DG day tank.	After each run of ≥1 hour
TR 16.8.3.7	Perform DG inspection, during shutdown, in accordance with manufacturer's recommendations for this class of standby service.	18 months
TR 16.8.3.8	Verify that the fuel oil transfer pump transfers fuel from each fuel storage tank to the day tank of each DG via the installed cross-connection lines.	18 months
TR 16.8.3.9	----------------------------------NOTE------------------------------ This Testing Requirement may be performed in conjunction with periodic pre-planned preventative maintenance activity that causes the DG to be inoperable provided that performance of the Testing Requirement does not increase the time the DG would be inoperable for the maintenance activity alone. --- Verify, during shutdown, that the turning gear engaged or emergency stop features prevent DG starting only when required.	18 months
TR 16.8.3.10	Perform a pressure test of those portions of the diesel fuel oil system designed to ASME Section III, subsection ND in accordance with applicable NRC-approved ASME code requirements.	10 years
TR 16.8.3.11	For each fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.	10 years
TR 16.8.3.12	Verify DG battery temperature is ≥45°F.	12 hours

BASES

The Testing Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides and Generic Letters referenced below.

TR 16.8.3.9 is modified with a note. This TR specifies that it is to be performed during shutdown. This note allows the TR to be performed during preplanned Preventative Maintenance (PM) activities that would result in the diesel generator being inoperable. This TR can be performed at that time as long as it does not increase the time the diesel generator is inoperable for the PM activity that is being performed. The note is only applicable at that time. The provision of the note shall not be utilized for operational convenience.

Since the McGuire emergency diesel generator manufacturer (Nordberg) is no longer in business, McGuire engineering is the designer of record. Therefore, in the absence of manufacturer recommendations, McGuire engineering will determine the appropriate actions required for nuclear class diesel service taking into account McGuire diesel generator maintenance and operating history and industry experience where applicable.

Draining of the DG fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at 10 year intervals by Regulatory Guide 1.137 (Ref. 7), paragraph 2.f. TR 16.8.3.11 also requires the performance of the ASME Code, Section XI (Ref. 8), examinations of the tanks. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This TR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this TR, provided that accumulated sediment is removed during performance of the Test.

The DG fuel oil storage tanks are currently deferred from the requirements of the Federal EPA regulations for underground storage tanks (USTs) on the basis that they are controlled through other programs and requirements. McGuire Environmental, Health and Safety group must be consulted regarding any changes to the testing requirements of these tanks to ensure Federal UST regulations are continued to be met.

Verification that the D/G Battery temperature is greater than or equal to 45°F will ensure sufficient battery capacity to perform its design function. This is based on the Diesel Generator Battery and Charger Sizing Calculation. Since a Battery Area Temperature indication is representative of the actual Diesel Generator Battery Temperature, use of this parameter is acceptable. However, if the temperature indication from this instrument is low (< 45°F) or the instrument is out of service, the actual D/G Battery Temperature must be determined. This may be accomplished by measuring the battery skin temperature, the battery electrolyte temperature, or the internal battery compartment temperature.

REFERENCES

1. Regulatory Guide 1.9, Selection of Diesel Generator Set Capacity for Standby Power Supplies, March 10, 1971.

2. Regulatory Guide 1.108, Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants, Revision 1, August 1977.

3. Regulatory Guide 1.137, Fuel-Oil Systems for Standby Diesel Generators, Revision 1, October 1979.

4. Generic Letter 84-15, which modified the testing frequencies specified in Regulatory Guide 1.108.

5. Generic Letter 93-05, which reduced the surveillance requirements for testing of Diesel Generators during power operation.

6. Generic Letter 94-01, which removed the accelerated testing and special reporting requirements for Emergency Diesel Generators.

7. Regulatory Guide 1.137.

8. ASME, Boiler and Pressure Vessel Code, Section XI.

9. McGuire Nuclear Station UFSAR, Section 18.2.4, Chemistry Control Program.

10. McGuire License Renewal Commitments MCS-1274-00-00-0016, Section 4.6, Chemistry Control Program..

11. MCC-1381.05-00-0195, The 125 VDC Diesel Generator Battery and Charger Sizing Calculation.

16.9 AUXILIARY SYSTEMS

16.9.1 Fire Suppression Water System

COMMITMENT The Fire Suppression Water System shall be OPERABLE with:

a. Fire suppression pump C and one other fire suppression pump, with their discharge aligned to the fire suppression header, and

b. An OPERABLE flow path capable of taking suction from Lake Norman and transferring water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrants, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per SLC 16.9.2 and 16.9.4.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fire suppression pumps A and B inoperable. <u>OR</u> Water supply to pumps A and B inoperable.	A.1 Restore one pump (A or B) and its associated water supply to OPERABLE status.	7 days
B. Fire suppression pump C inoperable.	B.1 Restore pump to OPERABLE status. <u>OR</u> B.2 Verify fire suppression pumps A and B and their water supplies are OPERABLE and at least one can be aligned to the blackout diesel generator.	7 days 7 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Primary automatic starting function for one or more required fire suppression pump(s) inoperable.	C.1 Verify secondary automatic starting function for each affected fire suppression pump is OPERABLE. OR C.2 Place at least one fire suppression pump in continuous operation.	Immediately Immediately
D. Secondary automatic starting function for one or more required fire suppression pump(s) inoperable.	D.1 Verify primary automatic starting function for each affected fire suppression pump is OPERABLE. OR D.2 Place at least one fire suppression pump in continuous operation.	Immediately Immediately
E. Both primary and secondary automatic starting functions for one or more required fire suppression pump(s) inoperable. OR Jockey pumps unable to maintain system header pressure.	E.1 Place at least one fire suppression pump in continuous operation.	Immediately
F. Fire Suppression Water System inoperable for reasons other than Condition A, B, C, D, or E.	F.1 Establish a backup fire suppression water system.	24 hours

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time not met.	G.1 Restore the system to OPERABLE status.	1 hour
	OR	
	G.2.1 Be in MODE 3.	7 hours
	AND	
	G.2.2 Be in MODE 4.	13 hours
	AND	
	G.2.3 Be in MODE 5.	37 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.1.1 Start each fire pump (A & B, or C) and operate for ≥ 15 minutes on recirculation flow..	15 days on a STAGGERED TEST BASIS
TR 16.9.1.2 Verify each manual, power operated, or automatic valve in flow path is in its correct position.	31 days
TR 16.9.1.3 Perform a system flush of the outside distribution loop and verify no flow blockage.	6 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.9.1.4	Cycle each testable valve in flow path through one complete cycle of full travel.	12 months
TR 16.9.1.5	Verify each automatic valve in the flow path actuates to its correct position in response to a simulated automatic actuation signal.	18 months
TR 16.9.1.6	Verify each pump develops $\geq$ 2500 gpm at a system pressure of $\geq$ 125 psig.	18 months
TR 16.9.1.7	Cycle each valve in flow path that is not testable during plant operation through one complete cycle of full travel.	18 months
TR 16.9.1.8	Verify each fire suppression pump starts automatically in response to a simulated automatic actuation signal.	18 months
TR 16.9.1.9	Perform a system flow test in accordance with NFPA Fire Protection Handbook, 14th ed., Section 11, Chapter 5.	3 years

BASES

The OPERABILITY of the Fire Suppression Systems 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 system, spray, and/or sprinklers, Halon, 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. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression. For McGuire Nuclear Station, fire pumps A and B serve as a backup for each other. Pump C is located separately with an independent dedicated power supply.

BASES (continued)

The Testing Requirements (TR) provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. Compliance with the testing requirements of SLC 16.9.1 ensures the main fire pumps meet all specified testing mandated by the 1978 National Fire Protection Association Code (Licensing Basis Code). Additional testing is conducted under the scope of TR 16.9.1.6 to gather pump operational data for the purpose of performance trending.

TR 16.9.1.7 requires cycling each valve in the flow path that is not testable during plant operation through one complete cycle of full travel. Although 1RF823 (Unit 1) and 1RF834 (Unit 2) are Containment Isolation check valves in the flow path, these valves are excluded from this testing requirement for the following reasons:

1. Check valves do not perform a sectionalizing control or isolation function.
2. 1RF823 and 1RF834 do not perform a dedicated fire protection system function.
3. NFPA 25 states that each control valve shall be operated through its full range and returned to its normal position. NFPA 25 recommends inspection of check valves internally to verify that all components operate properly, move freely, and are in good condition.
4. This exclusion is consistent with industry practices.
5. During Unit refueling outages, the Fire Suppression Water System including the check valves has been available for use.
6. Reactor Building fire hose stations are inspected every 36 months requiring opening hose valves, allowing flow through the check valves and verifying the fire protect system flow path.
7. The most common failure mode for these check valves will not affect the ability of the valve to open.

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. These corrective measures include unit shutdown if a backup fire suppression water system is not established as required.

Regulatory codes and standards mandate that the fire suppression water system has automatic starting function to preclude the necessity of manual operator action. The fire suppression pumps have dual auto-start design functions to meet this requirement. The primary auto-start control circuit (0RFLP5000) will start fire pumps at higher pressure setpoints than those associated with the secondary auto-start control circuits (0RYPS5010 for pump A, 0RYPS5020 for pump B and 0RYPS5030 for pump C). Either primary or secondary auto-start control circuit is fully capable of providing the required automatic starting function.

Since the requirement for fire suppression pump automatic starting functions is intended to provide a high level of system standby readiness, it is equally acceptable to place at least one pump in continuous operation if all automatic starting functions are inoperable. Likewise, if the fire suppression water system jockey pumps are unable to maintain system header pressure, it is acceptable to maintain system OPERABILITY by placing at least one pump in continuous operation.

BASES (continued)

This selected licensee commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1

2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D

3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B

4. McGuire Fire Protection Review, as revised

5. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C

6. Fire Protection System OP/1/A/6400/02A

7. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)

8. Fire Protection Handbook, 14th Edition, Published by the National Fire Protection Association, Chapter 5, Section 11

9. McGuire Nuclear Station UFSAR, Section 18.2.8, Fire Protection Program.

10. McGuire License Renewal Commitments MCS-1274.00-00-0016, Section 4.13, Fire Protection Program.

16.9 AUXILIARY SYSTEMS

16.9.2 Spray and/or Sprinkler Systems

COMMITMENT Spray and/or Sprinkler Systems in Table 16.9.2-1 shall be OPERABLE.

APPLICABILITY Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Spray and/or Sprinkler Systems inoperable in an area in which redundant systems or components could be damaged.	A.1 Establish a continuous fire watch with backup fire suppression equipment.	1 hour
B. One or more required Spray and/or Sprinkler Systems inoperable in areas other than Condition A.	B.1 Establish fire watch patrol.	1 hour AND Once per hour thereafter

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.2.1 Verify each manual, power operated, or automatic valve in flow path which is accessible during plant operation is in its correct position.	31 days

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.9.2.2	Cycle each testable valve in flow path through one complete cycle of full travel.	12 months
TR 16.9.2.3	Verify each automatic valve in the flow path actuates to its correct position in response to a simulated automatic Fire Detection signal.	18 months
TR 16.9.2.4	Cycle each valve in flow path that is not testable during plant operation through one complete cycle of full travel.	18 months
TR 16.9.2.5	Perform a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity.	18 months
TR 16.9.2.6	Perform a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.	18 months
TR 16.9.2.7	Verify each manual, power operated, or automatic valve in flow path which is not accessible during plant operation is in its correct position.	18 months
TR 16.9.2.8	Perform an air flow test through each open head spray/sprinkler header and verify each open head nozzle is unobstructed.	3 years

TABLE 16.9.2-1

Spray and Sprinkler Systems

Elevation	Building	Room No.	Equipment
Elevation 695 ft	Auxiliary	501	RHR Pump 1A
		500	RHR Pump 1B
		506	RHR Pump 2A
		507	RHR Pump 2B
		508	Corridor
Elevation 716 ft	Auxiliary	600	Aux. FW Pump Room - Unit 1
		649	Nuclear Service Water Pumps
		627	Centrifugal Charging Pump 1A
		630	Centrifugal Charging Pump 1B
		601	Aux. FW Pump Room - Unit 2
		634	Centrifugal Charging Pump 2A
		637	Centrifugal Charging Pump 2B
		648	Cable Shaft
Elevation 733 ft	Auxiliary	723	Component Cooling Pumps
		701	Battery Room Trench Area
Elevation 750 ft	Auxiliary	801	Cable Room – Unit 1
		801C	Cable Room – Unit 2
		806	Component Cooling Pumps
		803A	1ETA HVAC Equip Room
		805A	2ETA HVAC Equip Room
Elevation 725 ft	Reactor		Pipe Corridor
			Lower Containment Ventilation Filters
Elevation 738 ft	Reactor		Annulus

BASES

The OPERABILITY of the Fire Suppression Systems 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 system, spray, and/or sprinklers, Halon, 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. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Testing Requirements (TS) provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met.

TR 16.9.2.4 requires cycling each valve in the flow path that is not testable during plant operation through one complete cycle of full travel. Although 1RF823 (Unit 1) and 1RF834 (Unit 2) are Containment Isolation check valves in the flow path, these valves are excluded from this testing requirement for the following reasons:

1. Check valves do not perform a sectionalizing control or isolation function.
2. 1RF823 and 1RF834 do not perform a dedicated fire protection system function.
3. NFPA 25 states that each control valve shall be operated through it's full range and returned to it's normal position. NFPA 25 recommends inspection of check valves internally to verify that all components operate properly, move freely, and are in good condition.
4. This exclusion is consistent with industry practices.
5. During Unit refueling outages, the Fire Suppression Water System including the check valves has been available for use.
6. Reactor Building fire hose stations are inspected every 36 months requiring opening hose valves, allowing flow through the check valves and verifying the fire protect system flow path.
7. The most common failure mode for these check valves will not affect the ability of the valve to open.

This selected licensee commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1

2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D

3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B

4. McGuire Fire Protection Review, as revised

5. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C

6. MCFD-1599 1.0 through 3.01

7. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)

8. McGuire Nuclear Station UFSAR, Section 18.2.8, Fire Protection Program.

9. McGuire License Renewal Commitments MCS-1274.00-00-0016, Section 4.13, Fire Protection Program.

16.9 AUXILIARY SYSTEMS

16.9.3 Halon Systems

COMMITMENT The following Halon Systems shall be OPERABLE:

a. Elevation 716 ft. - Auxiliary Building

Room No.	Equipment
600B	Turbine Driven Aux. FW Pump - Unit 1
601B	Turbine Driven Aux. FW Pump - Unit 2

b. Elevation 733 ft. - Auxiliary Building

Room No.	Equipment
703-704	Diesel Generators - Unit 1
714-715	Diesel Generators - Unit 2

APPLICABILITY Whenever equipment protected by the Halon System is required to be OPERABLE.

REMEDIAL ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more required Halon Systems inoperable in an area in which redundant systems or components could be damaged.	A.1	Establish a continuous fire watch with backup fire suppression equipment.	1 hour
B.	One or more required Halon Systems inoperable in areas other than Condition A.	B.1	Establish fire watch patrol.	1 hour <u>AND</u> Once per hour thereafter

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9.3.1	Verify each manual, power operated, or automatic valve in flow path is in its correct position.	31 days
TR 16.9.3.2	Verify Halon storage tank weight ≥ 95% of full charge weight and pressure ≥ 90% of full charge pressure.	6 months
TR 16.9.3.3	Verify system actuates upon receipt of a simulated manual and automatic actuation signal and damper closure devices receive an actuation signal upon system operation.	18 months
TR 16.9.3.4	Perform a flow test through headers and nozzles to assure no blockage.	18 months

BASES

The OPERABILITY of the Fire Suppression Systems 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 system, spray, and/or sprinklers, Halon, 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 a Halon System becomes inoperable, compensatory actions are required to be taken in the affected areas until the inoperable equipment is restored to service.

For Rooms 704 (1B D/G), 714 (2A D/G) and 715 (2B D/G), Condition B is applied if the Halon system is declared inoperable since there is no impact to redundant systems and components.

For Room 703 (1A D/G), Condition A is applied if the Halon system is declared inoperable since there is impact on redundant equipment (1CA-42B power cable).

The Testing Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a UL or FM approved method.

The main bank (1 cylinder for the TDCA Pump Room, 8 cylinders for the D/G Room) or the reserve bank (1 cylinder for the TDCA Pump Room, 8 cylinders for the D/G Room) provides a sufficient quantity of halon to totally flood any of the TD CA Pump Rooms or Diesel Generator Rooms with the required design concentrations. Therefore, the Halon System is OPERABLE with the system aligned to either the main or the reserve bank of cylinders. The system is aligned to the main or reserve bank of cylinders by means of a local manual toggle switch.

TR 16.9.3.1 requires that valves in the flow path for the required halon systems be verified to be in their correct position. Although the selector valves and the cylinder valves are in the flow path, these valves are excluded from this testing requirement for the following reasons:

1. There is no visible means of determining valve position,
2. The valves are spring loaded piston actuators which fail closed and require halon discharge header pressure to open (Selector Valves Only),
3. There is no credible means to mis-position these valves other than actual actuation of the halon system,
4. These valves are an integral component of the actuation circuitry for the halon system, which is tested per TR 16.9.3.3, and
5. This exclusion is consistent with fire protection industry practices.

This selected licensee commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1
2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D
3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B
4. McGuire Fire Protection Review, as revised
5. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C
6. MCM-1206.07-35
7. MC-1599 4.0, MC-2599-4.0
8. MCEE-120.08.07
9. MCEE-120.16.07
10. MCEE-133-00.17

11. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)

16.9 AUXILIARY SYSTEMS

16.9.4 Fire Hose Stations

COMMITMENT The fire hose stations shown in Table 16.9.4-1 shall be OPERABLE.

APPLICABILITY Whenever equipment in areas protected by the fire hose stations is required to be OPERABLE.

REMEDIAL ACTIONS

--NOTES--

1. One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station.

2. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station.

3. Signs shall be mounted above the gate wye(s) to identify the proper hose to use.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more fire hose stations inoperable in an area in which the hose is the primary means of fire suppression.	A.1 Provide gated wye(s) on nearest OPERABLE hose station(s).	1 hour
B. One or more fire hose stations inoperable in an area in which the hose is not the primary means of fire suppression.	B.1 Provide gated wye(s) on nearest OPERABLE hose station(s).	24 hours

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9.4.1	Perform visual inspection of the fire hose stations, accessible during plant operations, to assure all required equipment is at the station and the fire hose shows no physical damage.	Quarterly or in accordance with the performance based criteria stated in the Bases
TR 16.9.4.2	Perform a visual inspection of the fire hose stations not accessible during plant operations to assure all required equipment is at the station and the fire hose shows no physical damage.	18 months
TR 16.9.4.3	Remove each fire hose for inspection and reracking.	18 months
TR 16.9.4.4	Inspect all fire hose gaskets and replace degraded gaskets in the couplings.	18 months
TR 16.9.4.5	Open each hose station valve partially to verify valve OPERABILITY and no flow blockage.	3 years
TR 16.9.4.6	Conduct a hose hydrostatic test at a pressure $\geq$ 150 psig or $\geq$ 50 psig above maximum fire main operating pressure, whichever is greater.	3 years

TABLE 16.9.4-1
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FIRE HOSE STATIONS

Number	**Location**	**Elevation (feet)**
157	55-FF	695
158	57-FF	695
175	51-LL/MM	716
176	55-MM	716
177	55-QQ	716
178	58/59-MM	716
179	61-LL	716
180	52-CC	716
181	54-C6	716
182	58-CG	716
183	59-CC /DD	716
167	51-JJ/KK	733
168	52-MM/NN	733
169	55-NN	733
170	57-LL	733
171	54-HH	733
172	58HH	733
173	60-MM/NN	733
174	61-JJ/KK	733
887	53-DD	733
889	51/52-DD	733
890	51-BB	733
891	40-CC	733
892	43/44-DD	733
893	40-AA/BB	733
894	44-AA/BB	733
895	46-BB	733
897	60-DD	733
898	61-BB	733
899	66-BB	733
900	68-AA/BB	733
901	72-BB	733
902	68/69-DD	733
903	72-DD	733
904	58-CC/DD	733

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FIRE HOSE STATIONS

Number	Location	Elevation (feet)
913	45-AA/BB	733
914	66BB	733
1184	56-JJ	733
161	50/51-MM	750
162	54/55-LL	750
163	54-JJ	750
164	56-QQ	750
165	58-LL/MM	750
166	61-MM	750
302	60-KK	750
303	52-GG	750
961	45-BB	750
962	46-CC	750
963	51-BB	750
964	51-BB	750
965	51-CC	750
966	56-DD	750
967	67-BB	750
968	66-CC	750
969	61-CC	750
970	61-BB	750
971	58-BB	750
972	57-DD	750
1185	58-JJ	750
184	54-KK	767
185	54-MM	767
186	50/51-MM	767
191	56-CG	767
192	58-JJ	767
193	60-MM	767
194	61/62-MM	767
974	51-BB	767
975	61-BB	767

BASES

The OPERABILITY of the Fire Suppression Systems 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 system, spray, and/or sprinklers, Halon, 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. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

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

The location of the required equipment at the fire hose station and the physical condition of fire hose is critical to fire brigade operations. The option of increasing or decreasing the frequency of the fire hose station inspections, based on hose performance, allows the ability to optimize plant resources. Should an adverse trend develop with fire hose station equipment or fire hose condition, the frequency of the inspection shall be increased. Similarly if the fire hose station equipment or fire hose condition trends are positive, the frequency of verification could be decreased. Through programmed trending of fire hose station inspections, fire hose stations will be maintained at predetermined reliability standards. The option to modify the frequency of SLC TR 16.9.4.1 is the responsibility of the Site Fire Protection Engineer via trending analysis of previous inspection results based on the following:

Annual review of the results of the completed fire hose station inspection procedures.

- If the results demonstrate that the fire hose stations are found acceptable at least 99% of the time over the 3 year rolling period, the frequency of conducting the fire hose station inspection may be decreased from - monthly to quarterly or - quarterly to semiannually or - semiannually to annually - as applicable. The frequency shall not be extended beyond annually (including grace period).

- If the results demonstrate that the fire hose stations are not found acceptable at least 99% of the time, the frequency of conducting the fire hose station inspections shall be increased from - annually to semiannually or - semiannually to quarterly or - quarterly to monthly - as applicable. The verification need not be conducted more often than monthly.

This commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions 2.C.(4) (Unit 1) and 2.C.(4) (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1

2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D

3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B

4. McGuire Fire Protection Review, as revised

5. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C

6. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition 2.C.(4).

7. McGuire Nuclear Station UFSAR, Section 18.2.8, Fire Protection Program.

8. McGuire License Renewal Commitments MCS-1274.00-00-0016, Section 4.13, Fire Protection Program.

16.9 AUXILIARY SYSTEMS

16.9.5 Fire Rated Assemblies

COMMITMENT

All fire rated assemblies (walls, floors/ceilings, cable tray enclosures and other fire barriers) separating:

a. Redundant analyzed Post Fire Safe Shutdown Equipment, or
b. Control Complex (i.e., Control Room, Cable Rooms and Battery Rooms) from the remainder of the plant, or
c. Safety related from non-safety related areas, or
d. Containment from non-containment areas,

AND

All sealing devices (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) in fire rated assembly penetrations shall be OPERABLE.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required fire rated assemblies and/or sealing devices inoperable.	A.1 Establish a continuous fire watch on at least one side of the affected assembly.	1 hour
	OR	
	A.2.1 Verify fire detectors on at least one side of the inoperable assembly are OPERABLE.	1 hour
	AND	
	A.2.2 Establish fire watch patrol.	1 hour AND Once per hour thereafter
	OR	
	A.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s)	Prior to terminating Required Action A1 or A2

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9.5.1	Verify that each unlocked fire door without electrical supervision is closed.	24 hours
TR 16.9.5.2	Verify that each fire door with automatic hold-open and release mechanisms are free of obstructions.	24 hours
TR 16.9.5.3	Verify that each locked closed fire door is closed.	7 days
TR 16.9.5.4	Perform a TADOT of the Fire Door Supervision System for each electrically supervised fire door.	31 days
TR 16.9.5.5	Perform an inspection of the automatic hold-open, release and closing mechanisms and latches on each associated fire door.	6 months
TR 16.9.5.6	Perform a functional test of the automatic hold-open, release and closing mechanisms and latches on each associated fire door.	18 months
TR 16.9.5.7	Perform a visual inspection of the exposed surfaces of each required fire rated assembly.	18 months
TR 16.9.5.8	------------------------------NOTE------------------------------ Samples shall be selected such that each damper will be inspected every 15 years. -- Perform a visual inspection of 10% of all required fire windows, fire dampers, and associated hardware.	18 months
TR 16.9.5.9	------------------------------NOTES------------------------------ 1. If a seal is found inoperable, an additional 10% of each type of sealed penetration shall be inspected until a 10% sample with no inoperable seals is found. 2. Samples shall be selected such that each penetration seal will be inspected every 15 years. -- Perform a visual inspection on 10% of each type of sealed penetration.	18 months

BASES

The functional integrity of the fire rated assemblies, including associated penetration seals, ensures that fires will be confined or adequately retarded so that the following criteria are achieved:

Fire will not damage redundant analyzed Post Fire Safe Shutdown equipment,
Fire will not spread from the balance of plant to the Control Complex,
Fire will not spread from non-safety related areas to safety related areas, and
Fire will not spread from non-containment areas to containment areas.

The fire related assemblies and associated penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire rated assemblies, including associated penetration seals (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) are considered operable when the visually observed condition is not degraded to a point that the assembly cannot perform its intended function. For fire rated assemblies that are questionable, an evaluation shall be performed, using the Problem Investigation Process, to determine the cause of any identified abnormal change in appearance or abnormal degradation and the effects of this change on the ability of the fire rated assembly to perform its function. Based on the results of the investigation process, additional assemblies may be selected for inspection.

During periods of time when a fire rated assembly is not OPERABLE, either: (A 1) a continuous fire watch is required to be maintained on at least one side of the affected barrier, or (A 2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established until the barrier is restored to OPERABLE status,or (A3) a licensee may choose to implement a different required action or combination of actions (e.g., additional administrative controls, operator briefings, temporary procedures, interim shutdown strategies, operator manual actions, temporary fire barriers, temporary detection or suppression systems). Such a change must be made to the approved Fire Protection Plan (FPP). However, the licensee must complete a documented evaluation of the impact of the proposed required action to the FPP. The evaluation must demonstrate that the required actions would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Any change to the FPP must maintain compliance with the GDC and 10CFR50.48(a) .

The evaluation of the required action should incorporate risk insights regarding the location, quantity, and type of combustible material in the fire area; the presence of ignition sources and their likelihood of occurrence; the automatic fire suppression and the fire detection capability in the fire area; the manual fire suppression capability in the fire area; and the human error probability where applicable.

The expectation is to promptly complete the corrective action at the first available opportunity and eliminate the reliance on the required action.

This Selected Licensee Commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1
2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D
3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B
4. McGuire Fire Protection Review, as revised
5. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C
6. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)
7. Door Schedules MC-1108-01, MC-1208-01-01, -02, -03, -04, -05 and -06.
8. Fire Plan Drawings MC-1384-07 series.
9. McGuire Nuclear Station UFSAR, Section 18.2.8, Fire Protection Program.
10. McGuire Licensing Renewal Commitments MCS-1274.00-00-0016, Section 4.13, Fire Protection Program.
11. NRC Regulatory Issue Summary 2005-07 - Compensatory Measures to Satisfy the Fire Protection Program Requirements, April 19, 2005

16.9 AUXILIARY SYSTEMS

16.9.6 Fire Detection Instrumentation

COMMITMENT The fire detection instrumentation for each fire detection zone shown in Table 16.9.6-1 shall be OPERABLE.

APPLICABILITY Whenever equipment protected by fire detection instrument is required to be OPERABLE.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more, but not more than half, of the Function A fire detectors in any zone inoperable.	A.1 Restore the inoperable instrument(s) to OPERABLE status.	14 days
B. More than half of the Function A fire detectors in any zone inoperable. OR One or more Function B fire detectors inoperable. OR Two or more adjacent fire detectors inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Establish fire watch patrol to inspect zones outside containment with inoperable instruments. AND B.2.1 Establish a fire watch patrol to inspect zones inside containment with inoperable instruments. OR B.2.2 Monitor containment air temperature at the locations given in ITS 3.6.5.1 or 3.6.5.2.	1 hour AND Once per hour thereafter 1 hour AND Once per 8 hours thereafter Once per hour

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more annulus fire detectors inoperable.	C.1 Perform a fire watch patrol of the annulus.	1 hour
	AND	
	C.2.1 Verify at least one adjacent annulus fire detector zone is OPERABLE.	Once per hour thereafter
	OR	
	C.2.2 Perform a fire watch patrol of the annulus if no adjacent zone is OPERABLE.	Once per 8 hours thereafter

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.6.1 Verify the non-supervised circuits associated with detector alarms between the instrument and the control room are OPERABLE.	31 days
TR 16.9.6.2 Verify the NFPA Standard 72D supervised circuits supervision associated with detector alarms are OPERABLE.	6 months
TR 16.9.6.3 Perform a TADOT on fire detectors which are accessible during plant operation.	6 months

(continued)

TESTING REQUIREMENTS (continued)

	TEST	FREQUENCY
TR 16.9.6.4	Perform a TADOT on fire detectors which are not accessible during plant operation.	Prior to entering MODE 4 when the unit has been in MODE 5 for 24 hours or more, if testing has not been performed in previous 6 months
TR 16.9.6.5	--------------------------------NOTE------------------------------ Different detectors shall be selected for each test. --- Perform a TADOT on at least one detector on each signal initiating circuit for fixed temperature/rate of rise restorable spot type heat detectors which are accessible during plant operation.	6 months
TR 16.9.6.6	Perform a TADOT on fixed temperature/rate of rise restorable spot type heat detectors which are not accessible during plant operation.	Prior to entering MODE 4 when the unit has been in MODE 5 for 24 hours or more, if testing has not been performed in previous 6 months
TR 16.9.6.7	--------------------------------NOTES------------------------------ 1. For each failure that occurs, two additional detectors shall be removed and tested. 2. Replacement of all fixed temperature/rate of rise non-restorable spot type heat detectors within population satisfies testing requirement. --- Perform a functional test on at least 2% of the fixed temperature/rate of rise non-restorable spot type heat detectors.	5 years
TR 16.9.6.8	Replace all fixed temperature/rate of rise non-restorable spot type heat detectors.	15 years

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION[1]

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function[2]
1	Reactor Coolant Pump 1A	RCP-1A	0	1	A
2	Reactor Coolant Pump 1B	RCP-1B	0	1	A
3	Reactor Coolant Pump 1C	RCP-1C	0	1	A
4	Reactor Coolant Pump 1D	RCP-1D	0	1	A
5	Reactor Coolant Pump 2A	RCP-2A	0	1	A
6	Reactor Coolant Pump 2B	RCP-2B	0	1	A
7	Reactor Coolant Pump 2C	RCP-2C	0	1	A
8	Reactor Coolant Pump 2D	RCP-2D	0	1	A
29	Aux. Bldg. Vent Filter	KK52-53 EL. 767	2	0	A
30	Elec. Pen. Rm.	CC-51 EL. 767	9	0	A
31	Elec. Pen. Rm.	CC-51 EL. 750	10	0	A
32	Elec. Pen. Rm.	CC-51 EL. 733	11	0	A
33	Unit 2 Aux. Bldg. Vent. Filter	KK-59/60 EL. 767	2	0	A
34	Unit 2 Elec. Penetration Room	CC-61 EL. 767	9	0	A
35	Unit 2 Elec. Penetration Room	CC-61 EL. 750	10	0	A
36	Unit 2 Elec. Penetration Room	CC-61 EL. 733	11	0	A
37	Diesel Gen. 1A	CC-43 EL. 733	0(0)	8(4)	A(B)
38	Diesel Gen. 2A	CC-69 EL. 733	0(0)	8(4)	A(B)
39	Cable Room	CC-55 EL. 750	6	5	A
40	Control Room	CC-56 EL. 767	24	19	A
41	Swgr. Rm. IETA	AA-49 EL. 750	9	0	A
42	Swgr. Rm. IETB	AA-49 EL. 733	10	2	A
43	SWG. Room 2ETA	AA-62 EL. 750	9	0	A
44	SWG. Room 2ETB	AA-62 EL. 733	10	2	A
45A	Battery Room EVCA	CC-54 EL. 733	2	2	A
45B	Battery Room EVCB	CC-55 EL. 733	2	2	A

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION[1]

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function[2]
45C	Battery Room EVCC	CC-56 EL. 733	2	2	A
45D	Battery Room EVCD	CC-57 EL. 733	2	2	A
45G	Battery Chg. Equip. & Pnl EVCA, EVCC	CC-56 EL. 733	13	0	A
45H	Battery Chg. Equip. & Pnl EVCB, EVCD	BB-56 EL. 733	12	0	A
50	Diesel Gen. 1B	BB-43 EL. 733	0(0)	8(4)	A(B)
51	Diesel Gen. 2B	BB-69 EL. 733	0(0)	8(4)	A(B)
52	Unit 2 Cable Room	CC-57 EL. 750	6	5	A
61	Cont. Spray Pump 1A/Corridor	GG-55 EL. 695	2	2	A
62	Cont. Spray Pump 1B/Cooridor	GG-56 EL. 695	2	2	A
63	RHR Pump 1B	FF-54 EL. 695	1	1	A
64	RHR Pump 1A	GG-54 EL. 695	1	1	A
66	Cont. Spray Pump 2B/Corridor	GG-56 EL. 695	2	2	A
67	Cont. Spray Pump 2A/Cooridor	GG-57 EL. 695	2	2	A
68	RHR Pump 2A	GG-58 EL. 695	1	1	A
69	RHR Pump 2B	FF-58 EL. 695	1	1	A
70	Aux. FW Pumps	BB-51 EL. 716	10(0)	80)	A(B)
72	Mech. Pen. Rm./Cables	JJ-51 EL. 716	4	4	A
73	Corridor/Cables	HH-53 EL. 716	5	5	A
74	Sample Panel/Cables	EE-55 EL. 716	5	5	A
75	Cent. Chg. Pump 1B	JJ-55 EL. 716	2	2	A
76	Cent. Chg. Pump 1A	JJ-55 EL. 716	2	2	A
77	PD Pump #1	JJ-54 EL. 716	2	2	A
78	Safety Injection Pump 1A	HH-54 EL. 716	2	2	A
79	Safety Injection Pump 1B	GG-54 EL. 716	2	2	A
80	Aisle/Cables	GG-55 EL. 716	12	12	A
81	Aisle/Cables	GG-57 EL. 716	10	10	A

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION(1)

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function(2)
82	Cent. Chg. Pump 2B	JJ-57 EL. 716	2	2	A
83	Cent. Chg. Pump 2A	JJ-57 EL. 716	2	2	A
84	PD Pump #2	JJ-58 EL. 716	2	2	A
85	Safety Injection Pump 2A	HH-58 EL. 716	2	2	A
86	Safety Injection Pump 2B	GG-58 EL. 716	2	2	A
87	Aux. FW Pumps	CC-60 EL. 716	10(0)	8(1)	A(B)
88	Mech. Penetration Room/Cables	JJ-61 EL. 716	4	4	A
90	Corridor/Cables	NN-59 EL. 716	5	5	A
91	Corridor/Cables	EE-53 EL. 733	4	4	A
92	Corridor/Cables	JJ-51 EL. 733	6	6	A
93	Corridor/Cables	NN-52 EL. 733	11	11	A
94	Aisle/Cables	JJ-55 EL. 733	9	9	A
95	600V MCC 1EMXB - 1EMXB3	FF-55 EL. 733	1	1	A
96	Cable Tray Access	EE-55 EL. 733	1	1	A
97	Cable Tray Access	EE-57 EL. 733	1	1	A
98	600V MCC 2EMXB - 2EMXB3	FF-57 EL. 733	1	1	A
99	Aisle/Cables	JJ-57 EL. 733	9	9	A
100	Corridor/Cables	NN-58 EL. 733	12	12	A
101	Corridor/Cables	JJ-61 EL. 733	6	6	A
102	Corridor/Cables	EE-59 EL. 733	4	4	A
103	Corridor/Cables	MM-51 EL. 750	6	6	A
104	Hatch Area Cables	LL-53 EL. 750	7	7	A
106	600V MCC 1EMXA	FF-54 EL. 750	2	2	A
107	600V MCC 2EMXA	FF-57 EL. 750	3	3	A
108	Aisle/Cables	JJ-55 EL. 750	14	14	A
109	Hatch Area Cables	PP-57 EL. 750	15	15	A
110	Corridor/Cables	PP-60 EL. 750	8	8	A
111	Corridor/Cables	LL-59 EL. 750	6	6	A

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION(1)

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function(2)
112	Aisle/Cables	JJ-57 EL. 750	13	13	A
113	HVAC Equipment Area/Cables	FF-56 EL. 767	8	8	A
114	Respiratory Equipment Room	GG-54 EL. 767	1	1	A
115	Corridor/Cables	JJ-54 EL. 767	13	13	A
116	HVAC Equipment Area/Cables	NN-52 EL. 767	7	7	A
120	Environmental Lab	PP-55 EL. 767	1	1	A
122	HVAC Equipment Area	NN-59 EL. 767	7	7	A
123	Corridor/Cables	JJ-57 EL. 767	14	14	A
125	Fuel Pool Area	NN-62 EL. 778+10	19	14	A
127	Fuel Pool Area	NN-50 EL. 731+ 6	18	14	A
128	Aisle/Cable	EE-57 EL. 716	5	5	A
129	600V MCC 2EMXH	KK-56 EL. 733	1	1	A
130	Cables/KF Pumps	PP-52 EL. 750	4	4	A
131	Respiratory	HH-56 EL. 767	5	5	A
134	RB Pipe Corridor-Unit 1	215° - 270°	0	5	A
135	RB Pipe Corridor-Unit 1	270°- 315°	0	5	A
136	RB Pipe Corridor-Unit 1	315° - 0°	0	6	A
137	RB Pipe Corridor-Unit 1	0° - 44°	0	4	A
138	RB Pipe Corridor-Unit 1	44° - 90°	0	4	A
139	RB Pipe Corridor-Unit 1	90° - 126°	0	4	A
140	RB Pipe Corridor-Unit 1	126° - 173°	0	7	A
141	RB Below Oper. Floor-Unit 1	329° - 349°	0	7	A
142	RB Below Oper. Floor-Unit 1	13° - 29°	0	4	A
143	RB Below Oper. Floor-Unit 1	34° - 51°	0	3	A
144	RB Below Oper. Floor-Unit 1	51° - 124°	0	13	A
145	RB Below Oper. Floor-Unit 1	124° - 143°	0	3	A
146	RB Below Oper. Floor-Unit 1	143° - 167°	0	8	A

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION(1)

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function(2)
147	RB Below Oper. Floor-Unit I	RCP - 1A Motor	0	5	A
148	RB Below Oper, Floor-Unit 1	RCP - 1 B Motor	0	2	A
149	RB Below Oper. Floor-Unit 1	RCP - 1C Motor	0	4	A
150	RB Below Oper. Floor-Unit 1	RCP - 1D Motor	0	5	A
151	RB Below Oper. Floor-Unit 1	Purge Filter Bed	0	2	A
152	RB Below Oper. Floor-Unit 1	170°-190°, R20'-R35'	0	2	A
c153	RB Annulus - Unit 1	293° - 331°	10	10	B
c154	RB Annulus - Unit 1	324° - 0°	4	4	B
c155	RB Annulus - Unit 1	0° - 50°	5	5	B
c156	RB Annulus - Unit 1	50° - 88°	4	4	B
c157	RB Annulus - Unit 1	88° - 123°	24	24	B
c158	RB Annulus - Unit 1	123° - 165°	22	22	B
c159	RB Annulus - Unit I	333° - 16°	13	13	B
c160	RB Annulus - Unit I	16° - 54°	23	23	B
c161	RB Annulus - Unit 1	122° - 180°	16	16	B
c162	RB Annulus - Unit 1	180° - 256°	14	13	B
163	Unit 2 RB Pipe Corridor	215° - 270°	0	4	A
164	Unit 2 RB Pipe Corridor	270°- 315°	0	5	A
165	Unit 2 RB Pipe Corridor	315° - 0°	0	6	A
166	Unit 2 RB Pipe Corridor	0° - 44°	0	4	A
167	Unit 2 RB Pipe Corridor	44° - 90°	0	4	A
168	Unit 2 RB Pipe Corridor	90° - 126°	0	4	A
169	Unit 2 RB Pipe Corridor	126° - 173°	0	7	A
170	Unit 2 RB Below Oper. Floor	329° - 347°	0	7	A
171	Unit 2 RB Below Oper. Floor	13° - 29°	0	4	A
172	Unit 2 RB Below Oper. Floor	34° - 51°	0	3	A
173	Unit 2 RB Below Oper. Floor	51° - 124°	0	13	A

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION(1)

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function(2)
174	Unit 2 RB Below Oper. Floor	124° - 143°	0	3	A
175	Unit 2 RB Below Oper. Floor	143° - 167°	0	8	A
176	Unit 2 RB Below Oper. Floor	RCP - 2A Motor	0	4	A
177	Unit 2 RB Below Oper. Floor	RCP – 2B Motor	0	3	A
178	Unit 2 RB Below Oper. Floor	RCP – 2C Motor	0	3	A
179	Unit 2 RB Below Oper. Floor	RCP – 2D Motor	0	5	A
180	Unit 2 RB Below Oper. Floor	Purge Filter Bed	0	2	A
181	Unit 2 RB Below Oper. Floor	170°-190°, R20'-R35'	0	2	A
d182	Unit 2 RB Annulus	293° - 331°	10	10	B
d183	Unit 2 RB Annulus	324° - 0°	4	4	B
d184	Unit 2 RB Annulus	0° - 50°	5	5	B
d185	Unit 2 RB Annulus	50° - 88°	4	4	B
d186	Unit 2 RB Annulus	88° - 123°	24	24	B
d187	Unit 2 RB Annulus	123° - 165°	22	22	B
d188	Unit 2 RB Annulus	333° - 16°	13	13	B
d189	Unit 2 RB Annulus	16° - 54°	23	23	B
d190	Unit 2 RB Annulus	122° - 180°	16	16	B
d191	Unit 2 RB Annulus	180° - 256°	13	13	B
197	Mech. Pen. Rm./UHI Valves	JJ-52 EL. 750	5	5	A
198	Mech. Pen. Rm./UHI Valves	JJ-60 EL. 750	5	5	A
206	Control Room Control Board	AA-56 EL. 767	20	5	A
c153A	RB Annulus - Unit 1 (Note 3)	0°-360° EL. 745	0	Note 5	B
c153B	RB Annulus - Unit 1 (Note 3)	0°-360° EL. 765	0	Note 5	B
c153C	RB Annulus - Unit 1 (Note 3)	0°-360° EL. 785	0	Note 5	B
c153D	RB Annulus - Unit 1 (Note 3)	0°-360° EL. 805	0	Note 5	B
c153E	RB Annulus - Unit 1 (Note 3)	0°-360° EL. 820	0	Note 5	B
c153F	RB Annulus - Unit 1 (Note 3)	0°-360° EL. 835	0	Note 5	B

TABLE 16.9.6-1

FIRE DETECTION INSTRUMENTATION(1)

Detector Zone	Description	Location	Number of Smoke Detectors	Number of Heat Detectors	Function(2)
d182A	RB Annulus – Unit 2 (Note 4)	0°-360° EL. 745	0	Note 5	B
d182B	RB Annulus – Unit 2 (Note 4)	0°-360° EL. 765	0	Note 5	B
d182C	RB Annulus – Unit 2 (Note 4)	0°-360° EL. 785	0	Note 5	B
d182D	RB Annulus – Unit 2 (Note 4)	0°-360° EL. 805	0	Note 5	B
d182E	RB Annulus – Unit 2 (Note 4)	0°-360° EL. 820	0	Note 5	B
d182F	RB Annulus – Unit 2 (Note 4)	0°-360° EL. 835	0	Note 5	B

NOTES:
1. The fire detection instruments located within containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

2. Function A: Early warning fire detection and notification only.
   Function B: Actuation of fire suppression system and early warning and notification.

3. Upon implementation of NSM MG-12106/00, zones 153 – 162 in RB Annulus – Unit 1 will be deleted and zones 153A – 153F will be active fire detection instrumentation.

4. Upon implementation of NSM MG-22106/00, zones 182 – 191 in RB Annulus – Unit 2 will be deleted and zones 182A – 182F will be active fire detection instrumentation.

5. The fire detection instruments located in the RB Annulus are restorable, cable-type sensors which cover the entire 360 degrees of the annulus at each subzone elevation.

BASES

Fire detection instrumentation is required to be operable at all times unless a complete evaluation has been made of the area protected by any particular instrument and all equipment in that area has been identified and determined not to be required operable. This evaluation would have to consider not only mechanical equipment in the area but all piping, tubing, and cables that transit through the area.

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge the extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program. An inoperable detector is defined as: a) a fire alarm with no actual fire or b) a trouble alarm.

Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for the Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABLE status. If fire detection capability is monitored by a local or remote panel, the fire watch patrol needs only check the panel to verify no loss in fire detection capability. Note that the MNS Fire Protection Safe Shutdown Review considers the annulus to be part of the containment building.

This selected licensee commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

TR 16.9.6.7, "Perform a functional test on at least 2% of the fixed temperature/rate of rise non-restorable spot type heat detectors," is satisfied by either testing within population or replacement of all heat detectors as per TR 16.9.6.8.

TR 16.9.6.8, "Replace all fixed temperature/rate of rise non-restorable spot type heat detectors," purpose is compliance with NFPA 72-2002 Table 10.4.2.2 Device 13 (d) 3. This is applicable to all containment fire zones (Unit 1 zones 134 – 152, Unit 2 zones 163 – 181).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1

2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D

3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B

4. McGuire Fire Protection Review, as revised

5. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C

6. NFPA Codes 72D and 72E

7. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)

16.9 AUXILIARY SYSTEMS - FIRE PROTECTION SYSTEMS

16.9.7 Standby Shutdown System

COMMITMENT The Standby Shutdown System (SSS) shall be operable.

APPLICABILITY MODES 1, 2, and 3.

REMEDIAL ACTIONS

---NOTE---

1. The SRO should ensure that security is notified 10 minutes prior to declaring the SSS inoperable. Immediately upon discovery of the SSS inoperability, Security must be notified to implement compensatory measures within 10 minutes of the discovery.

2. If inoperable SSS component is located inside containment, repairs shall be made at the first outage which permits containment access.

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----NOTE----- Not applicable to the SSS Diesel Generator or 24 V Battery Bank and Charger. ---------- A. One or more required SSS components identified in Table 16.9.7-1 inoperable.	A.1 Verify the OPERABILITY of fire detection and suppression systems in the associated areas identified in Table16.9.7-1. AND A.2 Restore the component to OPERABLE status.	1 hour 7 days
B. SSS Diesel Generator or 24 V Battery Bank and Charger inoperable.	B.1 Verify the OPERABILITY of fire detection and suppression systems in the associated areas identified in Table16.9.7-1. AND	1 hour (continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued).	B.2 Verify offsite power and one emergency diesel generator OPERABLE. <u>AND</u> B.3 Restore the component to OPERABLE status.	1hour 7days
C. Total Unidentified LEAKAGE, Identified LEAKAGE, and reactor coolant pump seal leakoff > 20 gpm. <u>OR</u> Total reactor coolant pump seal leakoff > 16.3 gpm. <u>OR</u> Any reactor coolant pump No. 1 seal leakoff > 4.0 gpm.	C.1 Declare the Standby Makeup Pump inoperable. <u>AND</u> C.2 Enter Condition A.	Immediately
D. Lake Norman level below 746 feet.	D.1 Verify the "C" Fire Suppression Pump is OPERABLE (Unit 1 only).	1 hour
E. Required Action A.2 and its associated Completion Time not met.	E.1 Prepare and submit a Special Report to the NRC outlining the cause of the inoperability, corrective actions taken, and plans for restoring the SSS to OPERABLE status.	30 days
F. Required Action B.3 and its associated Completion Time not met.	F.1 Prepare and submit a Special Report to the NRC outlining the extent of repairs required, schedule for completing repairs, and basis for continued operation.	14 days

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9.7.1	Verify total Identified LEAKAGE, Unidentified LEAKAGE, and reactor coolant pump seal leakoff are ≤ 20 gpm. AND Verify total reactor coolant pump seal leakoff ≤ 16.3 gpm. AND Verify each reactor coolant pump No. 1 seal leakoff ≤ 4.0 gpm.	72 hours
TR 16.9.7.2	Verify the requirements for spent fuel water level in Surveillance Requirement 3.7.13.1 are met and the boron concentration in the spent fuel storage pool is within the limits specified in the COLR. OR Verify the refueling water storage tank is capable of being aligned to the SSS standby makeup pump.	7 days
TR 16.9.7.3	Verify fuel oil level in the SSS diesel generator fuel storage tank is ≥ 4.0 ft.	31 days
TR 16.9.7.4	Verify the SSS diesel generator starts from ambient conditions and operates for ≥ 30 minutes at ≥ 700 kW.	31 days
TR 16.9.7.5	Verify fuel oil properties of new and stored fuel oil for the SSS diesel generator are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
TR 16.9.7.6	Verify the SSS diesel generator 24 V battery voltage is ≥ 24 volts.	31 days
TR 16.9.7.7	Perform a CHANNEL CHECK of the SSS Instrumentation as required by Table 16.9.7-2.	31 days
TR 16.9.7.8	Verify the electrolyte level of each SSS 250/125 V battery bank is above the plates.	31 days

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.9.7.9	Verify the total battery terminal voltage of each SSS 250/125 V battery bank is ≥ 258/129 V on float charge.	31 days
TR 16.9.7.10	Verify the average specific gravity of each SSS 250/125 V battery bank is ≥ 1.200.	92 days
TR 16.9.7.11	Verify the standby makeup pump's developed head and capacity is greater than or equal to that required by the Inservice Testing Supplemental Program.	92 days
TR 16.9.7.12	Verify the SSS diesel generator 24 V batteries and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months
TR 16.9.7.13	Verify SSS diesel generator 24 V battery to battery and terminal connections are clean, tight, and free of corrosion.	18 months
TR 16.9.7.14	Perform a CHANNEL CALIBRATION of the SSS Instrumentation as required by Table 16.9.7-2.	18 months
TR 16.9.7.15	Perform inspection of SSS diesel generator in accordance with procedures prepared in conjunction with manufacturer's recommendations for class of service.	18 months
TR 16.9.7.16	Verify the SSS 250/125 V batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months
TR 16.9.7.17	Verify the SSS 250/125 V battery to battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.	18 months
TR 16.9.7.18	Verify the "C" solenoid to valve SA48ABC can be deenergized to provide steam supply to the turbine driven auxiliary feedwater pump.	18 months
TR 16.9.7.19	Verify the CA Storage Tank level is ≥ 20 feet.	24 hours
TR 16.9.7.20	Verify Lake Norman level is ≥ 746 feet	24 hours

TABLE 16.9.7-1

STANDBY SHUTDOWN SYSTEM
FIRE DETECTION & SUPPRESSION SYSTEMS VERIFICATION[1]

INOPERABLE SSS COMPONENT	FIRE DETECTION & SUPPRESSION SYSTEMS LOCATION								
	EL 716 EE-KK	EL 733 EE-KK	EL 750 EE-KK	Control Room	Battery Room	Cable Rooms	Turbine Driven AFW Pump	Motor Driven AFW Pump	Containment
SSS Diesel Generator[3]	X	X	X	X	X	X	X	X	Note 2
SSS DG Starting 24 V Battery Bank and Charger[3]	X	X	X	X	X	X	X	X	Note 2
Standby Makeup Pump and Water Supply	X	X	X						
SSS 250/125V Battery and Charger[3]				X	X	X			Note 2
Turbine Driven AFW Pump and Water Supplies[4]								X	
Turbine Driven AFW Pump Solenoid "C"								X	
Groundwater Drainage Sump Pump A, Sump A AND Groundwater Drainage Sump Pump A, Sump B								X	
Fire Suppression Pump "C" (see Condition D).								X	
INSTRUMENTATION:									
1. RCS Pressure				X	X	X			Note 2
2. Pressurizer Level				X	X	X			Note 2
3. SG Level				X	X	X			Note 2
4. Incore Temperature				X	X	X			Note 2
5. NC Wide Range Cold Leg Temperature				X	X	X			Note 2

NOTES:

1. If fire detection and/or suppression systems are inoperable, then the ACTION statement(s) of the applicable fire detection and/or suppression SLC shall be complied with.
2. Monitor containment air temperature at least once per hour at the locations specified in Technical Specification Surveillance Requirement 3.6.5.1 or 3.6.5.2, in lieu of verification of operability of systems inside containment.
3. With this component inoperable, then denoted areas of both units are affected.
4. Water supplies include the Auxiliary Feedwater Storage Tank (CAST) and Condenser Circulating Water (RC) System via valves CA-161C and 162C. Continuous vents at 1/2RN-1065 and 1RN-1066 support OPERABILITY of the RC source for Unit 1 only.

TABLE 16.9.7-2

STANDBY SHUTDOWN SYSTEM
INSTRUMENTATION TESTING REQUIREMENTS

	INSTRUMENT	REQUIRED CHANNELS	TESTING REQUIREMENTS	READOUT LOCATION
1.	Reactor Coolant Pressure	1	TR 16.9.7.7 TR 16.9.7.14	SSF Control Panel
2.	Pressurizer Level	1	TR 16.9.7.7 TR 16.9.7.14	SSF Control Panel
3.	Steam Generator Level (Wide Range)	1 per SG	TR 16.9.7.7 TR 16.9.7.14	SSF Control Panel
4.	Incore Temperature	1	TR 16.9.7.7 TR 16.9.7.14	SSF Control Panel
5.	Standby Makeup Pump Flow	1	TR 16.9.7.14	SSF Control Panel
6.	NC Wide Range Cold Leg Temperature	2	TR 16.9.7.7 TR 16.9.7.14	SSF Control Panel

BASES

The Standby Shutdown System (SSS) is designed to mitigate the consequences of certain postulated fire incidents, sabotage, or station blackout events by providing capability to maintain HOT STANDBY conditions and by controlling and monitoring vital systems from locations external to the main control room. The facility is credited with the ability to cope with a station black out (SBO) event of 4 hour duration. This capability is consistent with the requirements of 10 CFR Part 50, Appendix R and 10 CFR 50.63.

By design, the SSS is intended to respond to those low-probability events which render both the control room and automatic safety systems inoperable. Because of the low probability of occurrence of these events, the remedial actions rely on compensatory action, timely repair or return to operability and, if necessary, a justification for continued operation.

Because the SSS performs a redundant fire protection function, compensatory action during periods when the SSS is inoperable relies largely on assurance of the operability of fire detection and suppression systems. Table 16.9.7-1 establishes requirements for operability of fire detection and suppression systems.

Both A&D NC Cold Leg Wide Range Temperatures are required for SSS operability. This conclusion is based on NRC Correspondence during issuance of the original operating license.

The Source Range Wide Range Neutron Flux Instrumentation was installed at the SSS Control Panel as part of NRC review of this system in the early 1980s. The indication is not required for SSS operability, based on the NRCs response to Duke dated July 21, 1983.

Controls and power to the pressurizer heater banks are included for SSF events; however, they are not required for SSS operability. NRC Generic Letter 86-10 provides that conclusion.

The Testing Requirements ensure that the SSS systems and components are capable of performing their intended functions. The testing requirements were based largely on SSS Technical Specifications for the Catawba Nuclear Station, which was approved prior to the issuance of the fuel load license for Unit 1 of that plant. Also considered in the formulation of the testing requirements were existing McGuire Technical Specifications, such as those for the 1E Diesel Generators, Refueling Water Storage Tank, Fire Protection & Detection Systems, and other Tech Specs which are related to the safe operation and/or shutdown of the plant.

The required level in the SSS diesel generator fuel storage tank ensures sufficient fuel for 3 ½ days of uninterrupted operation. Per Appendix R requirements, the unit must be in cold shutdown within 72 hours of going to the SSF. The 3 ½ day supply of fuel oil assures this capability.

Testing has demonstrated the ability of plant operations to start the SSF diesel within 10 minutes of the recognition of an SBO event, thus satisfying the intent of NUMARC 87-00 guidance. The SSF diesel generator has sufficient capacity and capability to operate equipment necessary to achieve and maintain safe shutdown conditions for a 4-hour SBO event.

BASES (continued)

Fuel oil for the SSS diesel generator is tested and maintained in accordance with the same Diesel Fuel Oil Testing Program used for the 4kV emergency diesel generators (see Technical Specification 5.5.13, Surveillance Requirement 3.8.3.2 and associated Bases).

Although the Standby Makeup Pump is not nuclear safety-related and was not designed according to ASME code requirements, it is tested quarterly to ensure its OPERABILITY. The Standby Makeup Pump (SMP) functions as part of the SSF to provide makeup capacity to the reactor coolant system and cooling flow to the reactor coolant pump (RCP) seals. The RCP seal leak-off flow is temperature dependent (i.e., the higher the temperature the higher the leak-off flow). During normal operation the RCP seals are supplied from the Centrifugal Charging Pump (CCP) drawing from the Volume Control Tank (VCT). During the SSF event, the SMP draws from the Spent Fuel Pool (SFP). During the SSF event there is no SFP cooling, so water injected into the RCP seals will have a higher temperature than during normal operation. The SMP is capable of providing a makeup capacity of 26 gpm. The revised SLC limit of 20 gpm total accumulative leakage is based on a calculation that was performed by Westinghouse, indicating increased RCP seal leak-off at higher seal water temperatures, to relate the SSF event leakage of 26 gpm at elevated RCP seal temperatures. This more conservative limit will ensure that the SMP will be capable of providing makeup and seal cooling flow equal to or greater than total leakage during the SSF event, increased RCP seal leak-off flow due to heat-up of the SFP, and still provide a margin of safety. As a conservative measure, during normal power operation the total accumulative system leakage (unidentified + identified + RCP seal leak-off flows) shall be limited to 20 gpm. The Testing Requirement concerning the SMP water supply ensures that an adequate water volume is available to supply the pump continuously for 72 hours.

The additional requirement that total RCP seal leak-off flow be $\leq$ 16.3 gpm resulted from a historical review of NRC correspondence that specified the SMP also provide for reactor coolant system makeup and boration in addition to RCP seal leakage requirements (Ref. 17). Calculations show that this upper limit for RCP seal leak-off provides sufficient margin to maintain the required unit conditions for a bounding SSS event.

Calculation MCC-1201.01-00-0053, Rev. 0, "MNS Units 1 & 2 Reactor Coolant Pump Response To Loss Of Seal Cooling," Sections 2 and 10 (Tab D, page 15) determined the elapsed time from loss of all seal cooling (loss of NV seal injection and loss of KC flow to the RCP thermal barrier heat exchanger) to when hot NC water entered the RCP No. 1 seal at varying seal leakoff rates. Chart interpolation determined that at a nominal No. 1 seal leakoff rate of 4 gpm, the seal would be at 235°F in 6.4 minutes from loss of all seal cooling event initiation. Therefore, for a maximum No. 1 seal leakoff of 4 gpm and if the Operators are instructed to stop all 4 RCPs at 3 minutes into the scenario, 3.4 minutes remain for the RCP motors to coast down to a stop and no seal rotation would occur above the No. 1 seal trip setpoint (235°F) during loss of all seal cooling. The 4 gpm limit is conservative based on the guidance provided in Westinghouse WCAP-17100, Section 1.2.3.4, "Response during a Loss of All Seal Cooling," and Westinghouse Technical Bulletin TB-04-22, Revision 1, for RCP coast down times and time for hot NC system water to reach the No. 1 seal on loss of all seal cooling.

The Groundwater Drainage Sump Pump A, in the A (Unit 1) and B (Unit 2) sumps, can be controlled and powered from the SSF. These Sump Pumps remove accumulation of groundwater, Turbine driven AFW Pump drains, and other miscellaneous sources. For the

BASES (continued)

SSS to be OPERABLE, a minimum of one of these pumps must be OPERABLE. Credit is taken for the groundwater underdrain system to transport water from one sump to the other.

The turbine driven AFW pump can be controlled from the SSF and is utilized during an SSS event to maintain adequate secondary side heat removal. For the SSS to be OPERABLE, the turbine driven AFW pump must be OPERABLE. The turbine driven AFW pump water supply for the 4 hour SBO event is provided by the CA Storage Tank (CAST). The water supply for the 72 hour fire event is initially provided by the CAST then later by manual alignment to the RC system via valves CA-161C and CA-162C. These valves are maintained closed and de-energized to prevent spurious actuation and air entrainment (Ref. 22). Adequate CAST inventory of 200,000 gallons (20 feet of level) is ensured by TR 16.9.7.19. For a fire event, an initial CAST inventory is needed to allow time to perform the manual alignment from the CAST to the RC system. For Unit 1 only, in the remote chance that the level of Lake Norman drops below 746 feet, air entrainment from the RC system cannot be prevented. In this case, adequate water supplies for a fire event are ensured by making up to the Unit 1 CAST from the Fire Suppression system using the "C" Fire Suppression pump.

The SSF is provided with its own 250/125 VDC power system which is independent from the normal 125 VDC and 120 VAC vital I&C power systems. The SSF batteries are charged by the SSF diesel generator and are available to power the SSF instruments and controls necessary to achieve and maintain hot standby conditions from the SSF control room following a station black out (SBO) event.

While the SSS 24 VDC battery charger is isolated for battery surveillance testing, the SSS Diesel Generator remains operable as long as the battery voltage is $\geq$ 24 volts.

The SSS 125V batteries and battery chargers consist of three pairs SDSP1, SDSP2 and SDSS. Each pair consists of a battery and associated battery charger. Pair SDSS can be used to substitute for either pair SDSP1 or SDSP2. Only two of these pairs are required operable since pair SDSS is spare.

This selected licensee commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 9.5.1

2. McGuire Nuclear Station SER Supplement 2, Chapter 9.5.1 and Appendix D

3. McGuire Nuclear Station SER Supplement 5, Chapter 9.5.1 and Appendix B

4. McGuire Nuclear Station SER Supplement 6, Chapter 9.5.1 and Appendix C

5. McGuire Fire Protection Review, as revised

6. McGuire Fire Protection Safe Shutdown Review

7. IEEE 308-1974, Class 1E Power Systems

8. IEEE 450-1975, Maintenance Testing & Replacement of Large Lead Storage Batteries

9. OP/O/B/6350/04, Standby Shutdown Facility Diesel Operation

10. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)

11. PIP 0-M-99-03926

12. PIP-M-01-3466

13. 10 CFR 50.63, Loss of All Alternating Current

14. Letter from H.B. Tucker to NRC, dated April 4, 1990, Requirements for Station Blackout.

15. Letter from H.B. Tucker to NRC, dated April 17, 1989, Requirements for Station Blackout.

16. McGuire Nuclear Station, Units 1 and 2, Safety Evaluation for Station Blackout (10CFR50.63), Dated February 19, 1992.

17. SAIC-91/1265, "Technical Evaluation Report, McGuire Nuclear Station, Station Blackout Evaluation," Dated December 10, 1991.

18. McGuire Nuclear Station UFSAR, Section 18.2.4, Chemistry Control Program.

19. MCS-1465.00-00-0019, "Plant Design Basis Specification For Station Blackout Rule," Rev. 3.

20. McGuire License Renewal Commitments MCS-1274.00-00-0016, Section 4.6, Chemistry Control Program.

21. PIP M-04-3317.

22. MCC-1223.42-00-0055, "Design Considerations and Bases for 1/2CA-161C and 1/2CA-162C Automatic Open Deletion Modifications MD101869 and MD201870."

16.9 AUXILIARY SYSTEMS

16.9.8 Groundwater Level Monitoring System

COMMITMENT

a. The groundwater level monitors listed in Table 16.9.8-1 shall be OPERABLE.

b. The groundwater level shall be maintained within the limits of Table 16.9.8-1

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more Reactor Building or Diesel Building Groundwater Level Monitors inoperable.	A.1	Restore the inoperable monitor to OPERABLE status.	7 days
B	Required Action and associated Completion Time of Condition A not met.	B.1	Provide an alternate method for determining the groundwater level for the monitored area.	Immediately
		<u>AND</u>		
		B.2	Enter the inoperable Level Monitor(s) into the Corrective Action Program.	24 hours
C.	Reactor Building or Diesel Building groundwater level not within limit.	C.1	Restore the groundwater level to within limit.	7 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Initiate an engineering evaluation to determine the cause and provide corrective action. <u>AND</u> D.2 Enter the groundwater level(s) not within limit into the Corrective Action Program.	Immediately 24 hours
E. Two Auxiliary Building Groundwater Level Monitors Inoperable.	E.1 Enter the inoperable level Monitor(s) into the Corrective Action Program. <u>AND</u> E.2 Restore at least one inoperable monitor to OPERABLE status. <u>OR</u> E.3 Provide an alternate method of determining the groundwater level for at least one of the affected monitored location(s).	24 hours 24 hours 24 hours
F. Groundwater level at three or more Auxiliary Building monitored locations not within limit.	F.1 Reduce the groundwater level to within limit.	1 hour

G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9.8.1	Verify the Auxiliary Building groundwater level within limits by absence of alarms, by visual observation of the monitor level gauge or by alternate methodology.	12 hours
TR 16.9.8.2	Perform a COT on the Auxiliary Building groundwater level monitors listed in Table16.9.8-1.	12 months

(continued)

TESTING REUIREMENTS (continued)

TEST		FREQUENCY
TR 16.9.8.3	Perform a CHANNEL CALIBRATION on the Auxiliary Building groundwater level monitors listed in Table 16.9.8-1.	12 months
TR 16.9.8.4	Perform a COT on the Reactor Building and Diesel Building groundwater level monitors listed in Table 16.9.8-1.	18 months
TR 16.9.8.5	Perform a CHANNEL CALIBRATION on the Reactor Building and Diesel Building groundwater level monitors listed in Table 16.9.8-1.	18 months

TABLE 16.9.8-1

GROUNDWATER LEVEL MONITORS

LOCATION and INSTRUMENT	EXTERIOR/INTERIOR	LEVEL LIMIT (MSL)	APPLICABILITY
Reactor Building 1WZLS5060	Exterior	731 ft	Unit 1
Diesel Building AA-40, ELEV. 736' 1WZLP5080	Interior	739 ft 2 in.	Unit 1
Diesel Building DD-42, ELEV. 736' 1WZLP5090	Interior	739 ft 2 in.	Unit 1
Reactor Building 2WZLS5060	Exterior	736 ft	Unit 2
Diesel Building BB-72, ELEV. 736' 2WZLP5080	Interior	739 ft 2 in.	Unit 2
Diesel Building DD-69, ELEV. 736' 2WZLP5090	Interior	739 ft 2 in.	Unit 2
Aux Bldg PP-51 1WZLP5100	Interior	731 ft	Unit 1 & 2
Aux Bldg QQ-56 0WZLP5110	Interior	731 ft	Unit 1 & 2
Aux Bldg PP-61 2WZLP5100	Interior	731 ft	Unit 1 & 2
Aux Bldg West Wall 1WZLS5070	Exterior	731 ft	Unit 1 & 2
Aux Bldg East Wall 2WZLS5070	Exterior	731 ft	Unit 1 & 2

BASES

The Reactor and Auxiliary Building complex for McGuire incorporates a permanent groundwater dewatering system that is designed to preclude groundwater from rising above a structural distress level of elevation 737 ft. Mean Sea Level (MSL) for the Auxiliary Building, or 23 feet below the site grade level of 760 ft. MSL.

The groundwater drainage system incorporates a grid system beneath the Reactor and Auxiliary Building basemats, 3 sumps in the Auxiliary Building, each with pumps and level alarms, a peripheral exterior drain system and 11 groundwater level monitors.

Originally, McGuire incorporated all 11 level monitors as Tech Spec monitors. Subsequently, an analysis performed by Design Engineering demonstrated that the Reactor and Diesel Generator Buildings were designed to withstand groundwater stresses up to 760 ft. MSL. Therefore, a Tech Spec revision was sought and obtained (Amendment Nos. 93 and 74) that removed all but 5 of the Auxiliary Building level monitors from the Tech Specs. The other 6 Reactor and Diesel Building level monitors were placed in Chapter 16 of the UFSAR as Selected Licensee Commitments (SLC). These 6 monitors, having locations listed in SLC Table16.9.8-1, were retained in administrative procedures for the groundwater monitoring program and utilized as an indication of any localized groundwater increases that may be indicative of increase due to ruptured pipes and as an indication of a problem with the underground grid system. This commitment was made as part of the justification for relaxing the groundwater monitoring Tech Spec.

The Reactor Building level monitors are exterior monitors and their first alarm is in the "Hi" alarm at 730 ft MSL on Unit 1 and 736 ft MSL on Unit 2. This ensures an alarm at or below the Groundwater Alert level (731 ft MSL on Unit 1, 736 ft MSL on Unit 2) is reached for the Reactor Building. The Diesel Building level monitors have similar first alarms but are termed "Alert" alarms and should not be confused with " Alert Levels." These alert levels for the Reactor Buildings have no safety significance because the Reactor and Diesel Buildings are analyzed for hydrostatic loads up to grade (760 ft.) elevation. Under the requirements of this SLC, if one or more Reactor or Diesel Building groundwater level monitors becomes inoperable or not within its level limit for 7 days or more, the condition shall be entered into McGuire's 10 CFR 50, Appendix B, Criterion XVI program (Corrective Action Program) for cause evaluation, corrective action, and trending.

The Auxiliary Building level monitors were placed in SLC 16.9.8 when McGuire converted to Improved Tech Specs. The SLC limits for the Auxiliary Building are provided to ensure that groundwater levels will be monitored and prevented from rising to the potential failure limit for the McGuire Units 1 and 2 Auxiliary Buildings. This potential failure limit is based on engineering calculations that have determined that the Auxiliary Buildings are susceptible to overturning due to buoyancy at elevation 737 feet Mean Sea Level (MSL). Under the requirements of this SLC, if groundwater level exceeds elevation 731 feet MSL, (3 out of 5 SLC groundwater monitor alarms), and cannot be reduced in one (1) hour, McGuire must begin reducing Units 1 and 2 to Mode 5, Cold Shutdown.

Elevation 731 feet MSL is the action level of the five Auxiliary Building groundwater monitors listed in Table16.9.8-1. The East Wall exterior monitor alarm at elevation 731 feet MSL is the Alert alarm. The other four (4) monitors are Hi-Hi alarms at elevation 731 feet MSL.

BASES (continued)

The East Wall exterior monitor was originally on the exterior of the Unit 2 Auxiliary Building and subsequently was enclosed by the construction of the Equipment Staging Building.

As required by Operations procedures, any alarms on SLC groundwater monitors will also be investigated. Additionally, if three (3) out of the five (5) groundwater monitors alarm at levels below the action levels, Operations will contact Civil Engineering for investigation and resolution of the increased groundwater level.

If one or more of the 5 Auxiliary Building groundwater monitors is determined to be inoperable, the monitor(s) will be considered to be indicating above the 731'-0" MSL until repaired and returned to an operable status or groundwater levels at the affected location(s) are determined to be within limits through alternate methods. Appropriate techniques shall be utilized to assure the accuracy of measurements taken through these alternate methods.

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 2.4.13.

2. McGuire Nuclear Station UFSAR, Appendix 2B.

3. McGuire Nuclear Station UFSAR, Chapter 9.5.8.

4. McGuire Nuclear Station UFSAR, Appendix 2D, Chapter 5.1.1.

5. McGuire Nuclear Station UFSAR , Chapter 7.6.11.

6. McGuire Nuclear Station UFSAR, Figure 2.4.13-1.

7. OP/1/A/6100/10I, Annunciator Response to Panel IAD-8

8. McGuire Nuclear Station SER Section 2.4.5 dated March 1, 1978.

9. PIP M-03-1377

10. PIP M-07-1139

11. NRC SER dated March 2, 1989 pursuant to McGuire License Amendment Request of January 27, 1988.

16.9 AUXILIARY SYSTEMS

16.9.9 Boration Systems – Flow Path (Operating)

COMMITMENT Two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from a boric acid tank via a boric acid transfer pump and a charging pump to the reactor coolant system, and

b. Two flow paths from the refueling water storage tank via charging pumps to the reactor coolant system.

Note: An OPERABLE charging pump used to satisfy OPERABILITY requirements of one boration flow path may not be used to satisfy OPERABILITY requirements for a second boration flow path.

APPLICABILITY MODES 1, 2, and 3.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required boron injection flow path inoperable.	A.1 Restore the required boron injection flow path to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Borate to the SDM requirements of Tech Spec 3.1.1. <u>AND</u> B.3 Restore the required boron injection flow path to OPERABLE status.	6 hours 6 hours 7 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 4.	30 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.9.1 Verify the temperature of piping associated with the flow path from the boric acid storage tanks is ≥ 65°F when it is a required water source	7 days
TR 16.9.9.2 Verify that each manual, power operated, or automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
TR 16.9.9.3 Verify that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.	18 months
TR 16.9.9.4 Verify that each charging pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
TR 16.9.9.5 Verify that the flow path from the boric acid tanks via a boric acid transfer pump and a charging pump delivers ≥ 30 gpm to the reactor coolant system.	18 months

BASES

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

In Modes 1, 2, and 3, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths

BASES (continued)

inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions. Further discussion is provided in Bases for Shutdown Margin Requirements (Tech Spec 3.1.1 and 3.1.2).

REFERENCES

1. PIP M-07-03237
2. DPND-1552.63-0099, Rev. 0, "Required Boration Flow Paths in Mode 4"
3. NRC Issuance of Amendments 184/166, Improved Technical Specification conversion and relocations to SLC Manual.

16.9 AUXILIARY SYSTEMS

16.9.10 Boration Systems – Charging Pumps (Operating)

(DELETED - COMBINED WITH 16.9.9)

16.9 AUXILIARY SYSTEMS

16.9.11 Borated Water Sources (Operating)

COMMITMENT As a minimum, the following borated water source(s) shall be OPERABLE as required by SLC 16.9.9:

a. A boric acid tank (BAT) and,

b. The refueling water storage tank.

APPLICABILITY MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 300°F.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required boric acid tank inoperable.	A.1 Restore the required boric acid tank to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Borate to the SDM requirements of TS 3.1.1 AND B.3 Restore the required boric acid tank to OPERABLE status.	6 hours 6 hours 7 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 4 with any RCS cold leg temperature $\leq$ 300°F.	30 hours

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Refueling water storage tank inoperable.	D.1 Enter the applicable Conditions and Required Actions of LCO 3.5.4, "Refueling Water Storage Tank."	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.11.1 Verify the refueling water storage tank solution temperature is ≥ 70°F and ≤ 100°F when the outside air temperature is < 70°F or > 100°F.	24 hours
TR 16.9.11.2 Verify the boron concentration of the required borated water source is within the limits specified in the COLR.	7 days
TR 16.9.11.3 Verify the borated water volume of the required borated water source is within the limits specified in the COLR.	7 days
TR 16.9.11.4 Verify the boric acid tank solution temperature is ≥ 65°F when the boric acid storage tank is a required source.	7 days

BASES

The borated water sources ensure that negative reactivity control is available during each mode of facility operation.

In Modes 1-3 and Mode 4 with all RCS cold leg temperatures above 300 °F, a minimum of two borated water sources are required to ensure single functional capability in the event an assumed failure renders one of the sources inoperable. The boration capability of either borated water source, in association with a flow path and charging pump, is sufficient to provide a SDM from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown.

The SLC commitment values are presented in the Core Operating Limits Report (COLR) as: (1) the minimum boron concentrations and minimum volumes necessary to attain and

BASES (continued)

maintain SDM in the BAT or the refueling water storage tank, (2) the minimum contained volumes in the BAT or the refueling water storage tank, and (3) a curve specifying the minimum contained volume in the BAT near EOC. The minimum contained water volume is based on the required volume to maintain shutdown margin, an allowance for water not available because of discharge line location and additional margin. The additional margin term includes allowances for instrument uncertainty, vortexing and a margin term consisting of at least 5% of the volume necessary for SDM. The COLR specified volumes are volumes reserved for use during a cooldown, and in conjunction with the boron concentrations, satisfy SDM requirements during Modes 1-3 and Mode 4 with all RCS cold leg temperatures above 300 °F.

Boric Acid Tank Requirements for Maintaining SDM

Required volume for maintaining SDM	Presented in the COLR
Unusable volume (to maintain full suction pipe)	4,199 gallons
Additional margin	4,100 gallons

Refueling Water Storage Tank Requirements for Maintaining SDM

Required volume for maintaining SDM	Presented in the COLR
Unusable volume (to maintain full suction pipe)	16,000 gallons
Additional margin	23,500 gallons

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

REFERENCES

None

16.9 AUXILIARY SYSTEMS

16.9.12 Boration Systems – Flow Path (Shutdown)

COMMITMENT

As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered by an emergency power source:

a. a flow path from a boric acid tank via a boric acid transfer pump and a charging pump to the reactor coolant system if the boric acid storage tank in SLC 16.9.14 is OPERABLE, or

b. the flow path from the refueling water storage tank via a charging pump to the reactor coolant system if the refueling water storage tank in SLC 16.9.14 is OPERABLE.

Note: An OPERABLE safety injection pump (and associated suction from RWST and discharge flowpath to cold legs) may be used in lieu of the charging pump in (b.) during Modes 5 and 6 when seal injection is not needed.

APPLICABILITY

MODES 4, 5, and 6.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required boron injection flow path inoperable. OR Required boron injection flow path not capable of being powered from an emergency power source.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend positive reactivity additions.	Immediately Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.12.1 Verify the temperature of piping associated with the flow path is $\geq$ 65°F when a flow path from the boric acid storage tank is used.	7 days
TR 16.9.12.2 Verify that the charging pump's or safety injection pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
TR 16.9.12.3 Verify that each manual, power operated, or automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

BASES

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water source, (2) charging pump, (3) separate flow path, (4) boric acid transfer pump, and (5) an emergency power supply from OPERABLE diesel generator. A safety injection pump with suction flow path from RWST and discharge flow path to RCS cold legs may also be used to perform boron injection functions during Modes 5 and 6.

In Modes 4, 5, and 6, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable. Further discussion is provided in Bases for Shutdown Margin Requirements (Tech Spec 3.1.1 and 3.1.2). The Mode 4 shutdown margin requirements are mandated by Tech Spec Section 3.1.1.

The capability was added to utilize a boration flow path from the RWST to the RCS cold legs via safety injection pump during Modes 5 and 6 as sufficient head is developed to borate the RCS at the LTOP actuation setpoint and below the applicable pressure limits of Tech Spec 3.4.2 (RCS P-T Limits).

For automatic valves and power operated valves which are OPERABLE and have an OPERABLE emergency power source, these valves may be repositioned as required to support other plant operations if the valves will move to their proper position on demand to establish the Boration Flow Path.

The REMEDIAL ACTION statement requires suspension of all operations 'involving CORE ALTERATIONS or positive reactivity changes.' The intent is that specific evolutions or operations that involve positive reactivity changes (fuel movement, dilutions, control rod movements or sustained NC temperature changes adding positive reactivity) are

BASES (continued)

discontinued if the conditions described above do not exist. There are operations (e.g., swapping ND trains, swapping KC trains, some testing) that can result in temperature oscillations that have insignificant effects on shutdown margin and can continue.

Operational or testing activities that result in NC temperature swings of 20 degrees F about an initial value have been judged not to constitute positive reactivity changes as described in this SLC when in MODE 5. There must be at least 500 ppm boron beyond the required Shutdown Boron Concentration for this interpretation to remain valid. This interpretation should not be used to establish sustained NC system heatups or cooldowns that result in sustained positive reactivity additions.

Limited Boron concentration changes are allowed for inventory control or testing provided SDM is maintained and Keff is <0.99. Operations are not permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM boron requirements.

REFERENCES

1. NSD-403: NGD Shutdown Risk Management
2. MSD-403: McGuire Shutdown Risk Management
3. Nuclear/Reactor Engineering Memo to File R.F.4.0.i, August 23, 1994 'NC Temperature Swings affect on Shutdown Margin'
4. PIPs M97-0601, M98-4643, M-07-03237
5. DPND-1552.63-0099, Rev. 0, "Required Boration Flow Paths in Mode 4"
6. NRC Issuance of Amendments 184/166, Improved Technical Specifications conversion and relocations to SLC Manual

16.9 AUXILIARY SYSTEMS

16.9.13 Boration Systems – Charging Pumps (Shutdown)

(DELETED-COMBINED WITH 16.9.12)

16.9 AUXILIARY SYSTEMS

16.9.14 Borated Water Sources (Shutdown)

COMMITMENT One of the following borated water sources shall be OPERABLE:

a. A boric acid tank (BAT), or

b. The refueling water storage tank.

APPLICABILITY MODE 4 with any RCS cold leg temperature ≤ 300°F,
MODES 5 and 6.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required borated water source inoperable.	A.1 Suspend CORE ALTERATIONS. <u>AND</u> A.2 Suspend positive reactivity additions.	Immediately Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.14.1 Verify the refueling water storage tank solution temperature is ≥ 70°F when the outside air temperature is < 70°F.	24 hours
TR 16.9.14.2 Verify the boron concentration of the required borated water source is within the limits specified in the COLR.	7 days

(continued)

TESTING REQUIREMENTS (continued)

TEST	FREQUENCY
TR 16.9.14.3 Verify the borated water volume of the required borated water source is within the limits specified in the COLR.	7 days
TR 16.9.14.4 Verify the boric acid tank solution temperature is $\geq$ 65°F when the boric acid storage tank is a required source.	7 days

BASES

The borated water sources ensure that negative reactivity control is available during each mode of facility operation.

In Mode 4 with any RCS cold leg temperature below 300 °F. and in Modes 5 and 6, one borated water source is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting core alterations and positive reactivity changes in the event the single borated water source becomes inoperable. The boration capability of one borated water source, in association with a flow path and charging pump, is sufficient to provide SDM of 1.3% delta k/k in Mode 4 and 1.0% delta k/k in Modes 5 and 6 after xenon decay and cooldown to 68° F.

The SLC commitment values are presented in the Core Operating Limits Report (COLR) as: (1) the minimum boron concentrations and minimum volumes necessary to attain and maintain SDM in the boric acid tank or the refueling water storage tank, (2) the minimum contained volumes in the boric acid tank or the refueling water storage tank, and (3) a curve specifying the minimum contained volume in the boric acid tank near EOC. The minimum contained water volume is based on the required volume to maintain shutdown margin, an allowance for water not available because of discharge line location and additional margin. The additional margin term includes allowances for instrument uncertainty, vortexing and a margin term consisting of at least 5% of the volume necessary for SDM. The COLR specified volumes and boron concentrations satisfy SDM requirements during Mode 4 with any RCS cold leg temperature below 300 °F and in Modes 5 and 6.

Boric Acid Tank Requirements for Maintaining SDM

Required volume for maintaining SDM	Presented in the COLR
Unusable volume (to maintain full suction pipe)	4,199 gallons
Additional margin	4,100 gallons

Refueling Water Storage Tank Requirements for Maintaining SDM

Required volume for maintaining SDM	Presented in the COLR
Unusable volume (to maintain full suction pipe)	16,000 gallons
Additional margin	23,500 gallons

BASES (continued)

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

REFERENCES

None

16.9 AUXILIARY SYSTEMS

16.9.15 Snubbers

COMMITMENT All snubbers shall be OPERABLE.

--------------------------------------NOTE-------------------------------------
Snubbers installed on non-safety systems may be excluded from these requirements provided their failure or the failure of the system on which they are installed would not have an adverse affect on any safety-related system.

APPLICABILITY At all times for snubbers located on systems required OPERABLE.

REMEDIAL ACTIONS

---NOTE---
Conditions A, B, and C are applicable to "seismic snubbers" as defined in the BASES.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE------------ If the opposite train of the associated system becomes inoperable for reasons not related to snubbers while in Condition A, exit Condition A and enter Condition C. ------------------------------ One or more seismic snubbers associated with one train of a multiple train system inoperable for maintenance or testing and the opposite train of the associated system is operable. OR One or more seismic snubbers associated with a single train system inoperable for maintenance or testing.	A.1.1	Verify that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method, is OPERABLE.	Immediately
		AND	
	A.1.2	Verify the opposite train of the associated system is operable, if a multiple train system.	Immediately
		AND	
	A.1.3	Log the affected system(s) for tracking in TSAIL.	Immediately
		AND	
	A.1.4	Enter the applicable ACTIONS for the train or component associated with the inoperable snubber(s).	72 hours
	OR		
	A.2	Declare the supported system inoperable.	Immediately

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more seismic snubbers associated with more than one train of a multiple train system inoperable for maintenance or testing.	B.1.1	Verify that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method, is OPERABLE.	Immediately
		AND	
	B.1.2	Enter the applicable ACTIONS for the train or component associated with the inoperable snubber.	12 hours
	OR		
	B.2	Declare the supported system inoperable.	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more seismic snubbers associated with one train of a multiple train system inoperable for maintenance or testing. AND The opposite train of the associated system is inoperable for reasons not related to snubbers.	C.1.1 Verify that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method, is OPERABLE.	Immediately
	AND	
	C.1.2 --------------Note------------- If the reason for the inoperability of either the snubber or the opposite train of the associated system is due to planned work, the risk assessment must be completed prior to starting work. --------------------------------- Initiate a qualitative risk assessment of the resulting configuration.	Immediately
	AND	
	C.1.3 Enter the applicable ACTION for the train or component associated with the inoperable snubber.	72 hours from failure to meet the COMMITMENT
	OR	
	C.2 Declare the supported system inoperable	Immediately
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the applicable ACTIONS for the train or component associated with the inoperable snubber.	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more snubbers with any significant non-seismic loads inoperable for maintenance or testing.	E.1 Enter the applicable ACTIONS for any affected system(s) and component(s) that are determined to be inoperable.	Immediately
	<u>OR</u>	
	E.2 Perform an engineering evaluation to determine the effect of the inoperable snubber on the operability of the associated system.	Prior to removing the snubber from service.
F. One or more snubbers inoperable due to actual failure of the snubber or failure to meet test acceptance criteria.	F.1 Perform an engineering evaluation to determine the effect of the inoperable snubber on the operability of the associated system.	72 hours

TESTING REQUIREMENTS

-----NOTES-----

1. Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

2. Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

3. As used herein, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

TEST		FREQUENCY
TR 16.9.15.1	-----NOTES----- 1. Snubbers are categorized as inaccessible or accessible during reactor operation and may be inspected independently according to the schedule determined by Table 16.9.15-1. 2. The first inspection interval using Table 16.9.15-1 shall be based upon the previous inspection interval as established by the requirements in effect before Technical Specification amendment 126. ----- Perform a visual inspection for each category of snubber.	In accordance with Table 16.9.15-1
TR 16.9.15.2	-----NOTE----- In case of a severe dynamic event, mechanical snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the mechanical snubbers have freedom of movement and are not frozen up ----- Perform an inspection to determine if there has been a severe dynamic event for systems which have the potential for a severe dynamic event.	18 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.9.15.3	------------------------------NOTE------------------------------ 1. The large bore steam generator hydraulic snubbers shall be treated as a separate population for functional test purposes and are functional tested under Sample Plan 1. 2. If testing continues under Sample Plan 2 to between 100-200 snubbers(or 1-2 weeks) and the accept region has not been reached, then the actual % of population quality (C/N) should be used to prepare for extended or 100% testing. -- Perform snubber functional testing on a representative sample of each type of snubber in accordance with one of the following three Sampling Plans: 1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or 2. Functionally test a sample size and determine sample acceptance or continue testing using Figure 16.9.15-1, or 3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.	18 months
TR 16.9.15.4	------------------------------NOTE------------------------------ The parts replacement shall be documented and the documentation shall be retained for the duration of the unit operating license. -- Verify that the service life of hydraulic snubbers has not been exceeded or will not be exceeded prior to the next scheduled surveillance inspection.	18 months

BASES

This commitment contains requirements for plant snubbers. There are snubbers installed for seismic loads only (i.e., those loads induced by seismic events, "earthquakes") and snubbers that are installed for the combined effects of both seismic loads and non-seismic loads (i.e., those dynamic loads induced by operational events such as steamhammer, waterhammer, LOCAs, and pipe rupture). Thus for the purpose of this commitment, there are two categories of snubbers:

1) snubbers which have only seismic loads, and snubbers which have both seismic and non-seismic loadings, but Engineering has determined that the non-seismic loads are insignificant and do not effect the operability of the associated system. Since the seismic loads are those of significance, these snubbers are termed "Seismic Snubbers" in this commitment; and

2) snubbers which have both seismic and non-seismic loadings and Engineering has determined that the non-seismic loads are significant and do effect the operability of the associated system.

The Remedial Actions for each of these snubber categories are discussed below. Remedial Action F.1 and the Testing Requirements of this commitment apply to both categories of snubbers. The programmatic requirements for the visual inspection and functional testing of snubbers do not meet the criteria in 10 CFR 50.36(c)(2)(ii) for inclusion in the plant TS, and as such, are appropriate for control by this commitment and are the same for both categories of snubbers.

The snubber requirements of SLC 16.9.15 were originally located in the Technical Specifications. The Nuclear Regulatory Commission (NRC) authorized the use of these requirements, while located in Technical Specifications, as an acceptable alternative to the requirements of the ASME Code, 1989 Edition, Section XI, Article IWF-5000 (References 3, 4). Any revision to these snubber visual inspection and functional test requirements shall consider the basis for the granted relief from the ASME Code requirements and any resulting requirement for NRC review and approval.

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. Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip, and 100 kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this specification would be of a different type, as would hydraulic snubbers from either manufacturer.

Snubbers removed from service for any reason cannot be considered OPERABLE since it is not connected to the supported system or component.

BASES (continued)

Seismic Snubbers

Seismic snubbers are installed primarily to address loads resulting from a seismic event. However, some seismic snubbers do have other non-seismic loads, but these other loads have been determined to have an insignificant effect on the operability of the associated system, as determined by Engineering. If used, TS LCO 3.0.8 contains the OPERABILITY requirements for seismic snubbers.

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated seismic snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more of these snubbers not being capable of performing their associated support function(s). Thus, any affected supported LCO(s) are not required to be declared not met solely for this reason, if risk is assessed and managed. This is appropriate because a limited length of time is allowed for inspection, testing, maintenance, or repair of one or more of these snubbers not capable of performing their associated support function(s), remedial actions are specified in this commitment, and the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function, and as applicable, due to the availability of the redundant train of the supported system.

If the allowed time expires and the seismic snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

Snubbers with Both Seismic and Significant Non-Seismic Loads

If the affected snubber has more than one function, one of which MUST be seismic loads, then LCO 3.0.8 may be applied. However, there must be a deterministic analysis that demonstrates that the supported system can still perform its function for the non-seismic load(s). For example, if the affected snubber has support functions for both seismic loads and LOCA loads (i.e., blowdown loads), then only that LOCA load is considered deterministically to determine if the system is OPERABLE. If the supported TS system is OPERABLE for the non-seismic loads, then LCO 3.0.8 may be applied to the seismic loads. Otherwise, LCO 3.0.8 may not be applied and the OPERABILITY requirements are contained in this commitment.

Remedial Actions - A

Remedial Action A applies when one or more seismic snubbers associated with one train of a multiple train system and the opposite train of the associated system is operable or associated with a single train system are inoperable for maintenance or testing, thus are not capable of providing their associated support function(s). This commitment allows up to 72 hours to restore the seismic snubber(s) before declaring the supported system inoperable, provided: 1) there is an immediate determination that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method (e.g., feed and bleed, firewater system or "aggressive secondary cooldown" using the steam generators) is OPERABLE, 2) the opposite

BASES (continued)

train of the supported system is OPERABLE, if applicable, and 3) the affected system is logged for tracking in TSAIL. The 72- hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the seismic snubber(s) are not capable of performing their associated support function, and due to the availability of the redundant train of the supported system (if applicable).

At the end of the specified 72-hour period the required seismic snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

Condition A is modified by a Note which specifies that if the opposite train of the associated system becomes inoperable for reasons not related to snubbers while in Condition A, Condition A can be exited and Condition C is entered.

If the provisions of LCO 3.0.8 are not entered, the supported system shall be declared inoperable immediately.

Remedial Action - B

When one or more seismic snubber(s) are not capable of providing their associated support function(s) to more than one train of a multiple train supported system, this commitment allows 12 hours to restore the seismic snubber(s) before declaring the supported system(s) inoperable, provided there is an immediate determination that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method (e.g., feed and bleed, firewater system or "aggressive secondary cooldown" using the steam generators) is OPERABLE. The 12-hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the seismic snubber(s) are not capable of performing their associated support function.

At the end of the specified 12-hour period the required seismic snubbers must be able to perform their associated support function(s), or the affected supported system(s) LCO(s) shall be declared not met.

If the provisions of LCO 3.0.8 are not entered, the supported system shall be declared inoperable immediately.

Remedial Action - C

When one or more seismic snubbers are not capable of providing their associated support function(s) to one train of a multiple train supported system, and the opposite train of the supported system is inoperable for reasons not related to snubbers, this commitment allows up to 72 hours to restore the seismic snubber(s) before declaring the supported system inoperable provided: 1) there is an immediate determination that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method (e.g., feed and bleed, firewater system or "aggressive secondary cooldown" using the steam generators) is OPERABLE, and 2)

BASES (continued)

there is an immediate assessment of risk associated with the resulting configuration and the risk assessment is acceptable. The 72-hour Completion Time from failure to meet the COMMITMENT (in case Condition C is entered after exiting Condition A) is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the seismic snubber(s) are not capable of performing their associated support function, and due to an acceptable conclusion of the risk assessment.

At the end of the specified 72-hour period the required seismic snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

If the provisions of LCO 3.0.8 are not entered, the supported system shall be declared inoperable immediately.

Risk Assessment and Management

Remedial Action A, B, and C require that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of these commitments should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule Process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. When using this commitment to remove seismic snubber(s) from an operable state, the risk assessment must ensure that at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or alternative core cooling method (e.g., feed and bleed, firewater system or "aggressive secondary cooldown" using the steam generators) is OPERABLE. This risk assessment is tracked by use of the TSAIL program. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. Actions that could be taken include protection of other trains or subsystems for example.

Remedial Action - D

If the Required Action and associated Completion Time of Condition A, B or C are not met, the applicable ACTIONS for the train(s) or components(s) associated with the inoperable seismic snubber shall be entered immediately.

Remedial Action - E

Should one or more snubbers which have any significant non-seismic loads be inoperable for the purposes of maintenance or testing. OPERABILITY of the affected system(s) and component(s) must be determined and the applicable ACTIONS entered immediately. If there remains a reasonable assurance of OPERABILITY of the affected system(s) or component(s) with the condition of an inoperable snubber(s), then it is not necessary to enter the respective ACTIONS for inoperable system(s) and component(s).

Remediate Action - F

BASES (continued)

Should one or more snubbers (of either category) fail to meet testing acceptance criteria or be discovered in a condition where failure is apparent, an engineering evaluation is to be performed within 72 hours, as described in "Functional Test Failure Analysis".

The snubber-testing program may remove snubbers from service and restore OPERABILITY of the snubber application by replacement with another like snubber. In this situation, if the removed snubber application by replacement with another like snubber. In this situation, if the removed snubber later fails to meet test acceptance criteria, Conditions A, B, C, and E are not applicable since the snubber component has no current required function; however, ACTION F.1 would be applicable. During the 72 hours allowed to perform an engineering evaluation, or at any other time when conditions of the affected system(s) and component(s) are determined to no longer support a reasonable assurance of OPERABILITY, applicable ACTIONS shall be entered immediately.

Visual Inspections

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. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE. A hydraulic snubber found with the fluid port uncovered and all hydraulic snubbers found connected to an inoperable common reservoir shall be classified as unacceptable and may be reclassified acceptable by functionally testing each snubber starting with the piston in the as-found setting, extending the piston rod in the tension direction.

Refueling Outage Inspections

At each refueling, the systems which have the potential for a severe dynamic event, specifically, the main steam system (upstream of the main steam isolation valves) the main steam safety and power-operated relief valves and piping, auxiliary feedwater system, main steam supply to the auxiliary feedwater pump turbine, and the letdown and charging portion of the NV system shall be inspected to determine if there has been a severe dynamic event.

In case of a severe dynamic event, mechanical snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the mechanical snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying

BASES (continued)

freedom of motion using one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; (3) 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 repaired before returning to power. The requirements of TR 16.9.15.1 are independent of the requirements of this item.

Functional Testing

During the first refueling shutdown and at least once per refueling thereafter, a representative sample of snubbers shall be tested using one of the following sample plans. The large bore steam generator hydraulic snubbers shall be treated as a separate population for functional test purposes and are functional tested under Sample Plan 1. A 10% random sample from previously untested snubbers shall be tested at least once per refueling outage until the entire population has been tested. This testing cycle shall then begin anew. For each large bore steam generator hydraulic snubber that does not meet the functional test acceptance criteria, at least 10% of the remaining population of untested snubbers for that testing cycle shall be tested. The sample plan shall be selected prior to
the test period and cannot be changed during the test period. The NRC shall be notified of the sample plan selected prior to the test period.

1. At least 10% of the required snubbers 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 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested; or

2. A representative sample of the required snubbers shall be functionally tested in accordance with Figure16.9.15-1. "C" is the total number of snubbers found not meeting the acceptance requirements (failures). The cumulative number of snubbers tested is denoted by "N." Test results shall be plotted sequentially in the order of sample assignment (i.e., each snubber shall be plotted by its order in the random sample assignments, not by the order of testing). If at any time the point plotted falls in the "Accept region, testing of snubbers may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers shall be tested until the point falls in the 'Accept" region, or all the required snubbers have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3. An initial representative sample of fifty-five (55) snubbers shall be functionally tested. For each snubber which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where
"C" is the number of snubbers found which do not meet the functional test acceptance criteria. This can be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber should be plotted as soon as it is tested. If the point plotted falls on or

below the "Accept" line, testing may be discontinued, If the point plotted falls above the "Accept" line, testing must continue unless all snubbers have been tested.

BASES (continued)

The representative samples for the functional test sample plans shall be randomly selected from the required snubbers and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of sizes, and capacities. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

Figure 16.9.15-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the NRC 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 at either the completion of their fabrication or at a subsequent date.

Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3. Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

BASES (continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be evaluated in a manner to ensure their OPERABILITY. This testing requirement shall be independent of the requirements stated in TR 16.9.15.3 for snubbers not meeting the functional acceptance criteria.

Service Life

The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be OPERABLE.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

REFERENCES

1. Letter from M. S. Tuckman to NRC, Licensing Position Regarding Snubbers, May 20, 1999.
2. Letter from NRC to H.B. Barron, Licensing Position Regarding Snubbers, July 9, 1999.
3. Letter from H.B. Barron to NRC, Request for Relief 97-005, Snubber Inspections - Performance and Schedule, December 17, 1997.
4. Letter from NRC to H.B. Barron, Relief Request for Snubber Visual examination and Functional Testing, May 27, 1998.
5. Letter from H.B. Barron to NRC, Request for Relief 01-004, June 1, 2001.
6. Letter from NRC to M.S Tuckman, Safety Evaluation of Relief Request No. 01-004, Alternative for Snubber Examinations, January 30, 2002.
7. Letter from G.R. Peterson to NRC, Request for Relief 03-002, March 8, 2004.
8. Letter from G.R. Peterson to NRC, RAI Response, September 22, 2004.
9. Letter from NRC to G.R. Peterson, Safety Evaluation of Relief Request No. 03-002, November 22, 2004.
10. Technical Specification Task Force (TSTF) 372-A, Revision 4, Addition of LCO 3.0.8. Inoperability of Snubbers.
11. TSTF-IG-05-03, Rev 1, Technical Specifications Task Force Implementation Guidance for TSTF-372-A, Revision 4, Addition of LCO 3.0.8, Inoperability of Snubbers.
12. Nuclear System Directive 415, Operational Risk Management (Modes 1-3) per 10 CFR 50.65(a)(4).
13. Federal Register, 70FR23252, Notice of Availability of Model Application Concerning Technical Specification Improvement to Modify Requirements Regarding the Addition of Limiting Condition for Operation 3.0.8 on the Inoperability of Snubbers Using the Consolidated Line Item Improvement Process.

TABLE 16.9.15-1

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1, 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extended Interval (Notes 3, 6)	Column A Repeat Interval (Notes 4, 6)	Column C Reduced Interval (Notes 5, 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
≥1000	29	56	109

NOTES:

1. The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. The categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

2. Interpolation between population or category size and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as described by interpolation.

3. If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

4. If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection shall be the same as the previous interval.

5. If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

6. The provisions of SLC 16.2.7 are applicable for all inspection intervals up to and including 48 months.

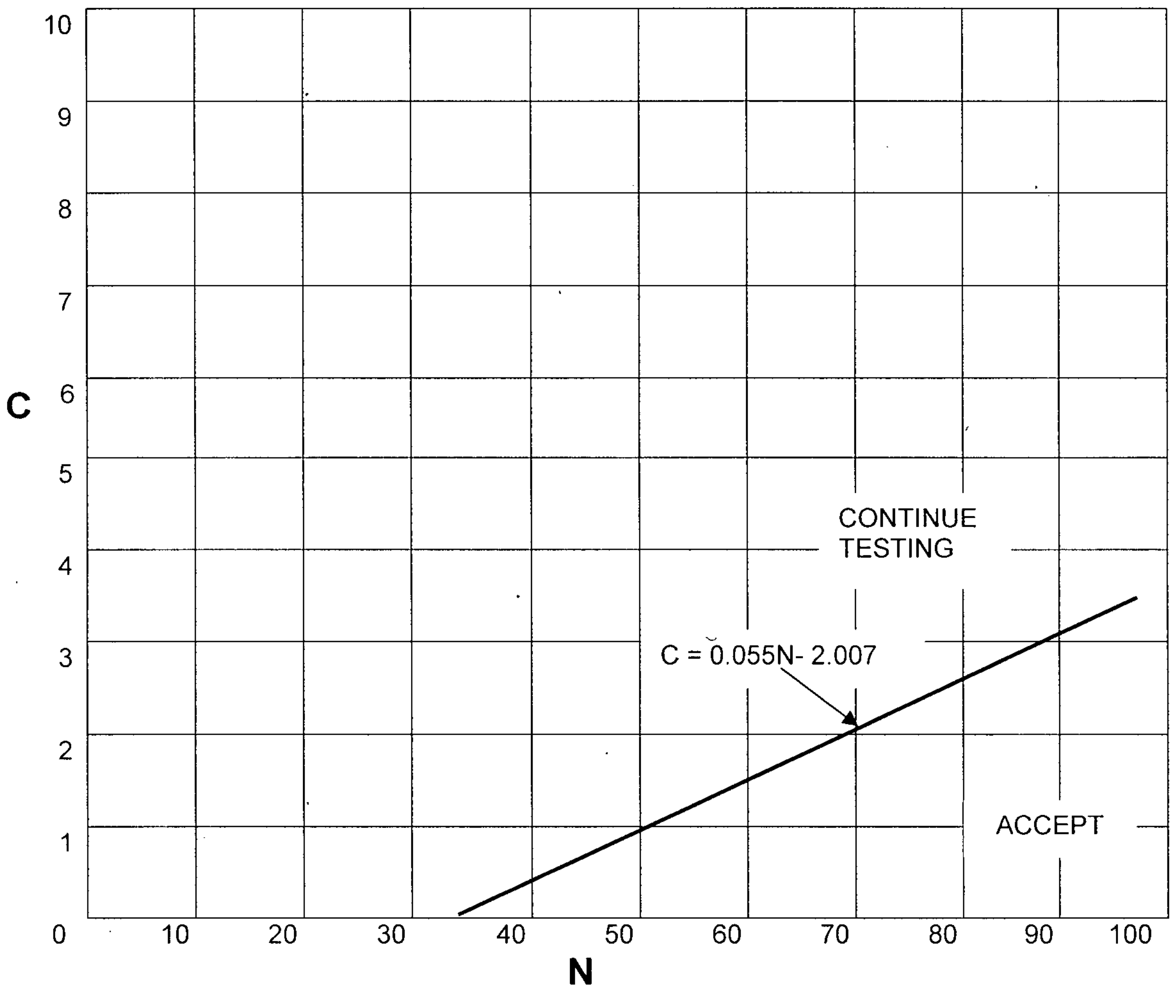


FIGURE 16.9.15-1

SAMPLE PLAN 2 FOR SNUBBER FUNCTIONAL TEST

16.9 AUXILIARY SYSTEMS

16.9.16 Area Temperature Monitoring

COMMITMENT The temperature of each area shown in Table 16.9.16-1 shall be maintained within the limits indicated in Table16.9.16-1.

APPLICABILITY Whenever the specified equipment in an affected area is required to be OPERABLE.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more areas, except Diesel Generator Rooms, exceeding temperature limit(s) shown in Table 16.9.16-1.	A.1 Initiate actions to restore the temperature within limit. AND A.2 Declare equipment in the affected area inoperable.	Immediately Immediately
B. Diesel Generator Room temperature is > 125°F.	B.1 Initiate actions to restore Diesel Generator Room temperature to within limits. AND B.2 Declare Diesel Generator inoperable.	Immediately Immediately
C. Diesel Generator Room temperature is < 55°F. AND Diesel Generator is not running.	C.1 Initiate actions to restore Diesel Generator Room temperature to within limits. AND C.2. Declare Diesel Generator inoperable.	Immediately Immediately

REMEDIAL ACTIONS (continued)

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.16.1 Verify temperature in each of the areas shown in Table 16.9.16-1 is within limits.	12 hours

TABLE 16.9.16-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1. Containment Spray Pump Rooms	145
2. Miscellaneous Terminal Cabinets	
a. TB208-209 (Turbine Building Unit 1)	150
b. TB496 (Fuel Building Unit 1)	150
c. TB1208-1209 (Turbine Building Unit 2)	150
d. TB1496 (Fuel Building Unit 2)	150
3. Residual Heat Removal Pump Rooms	145
4. Diesel Generator Rooms	$\geq$ 55 and $\leq$ 125
5. Spent Fuel Pool Cooling Pump Room	145

BASES

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The Diesel Generator Room temperature is measured by taking an average of the Battery Area and Control Panel Area thermocouples. This is indicated on an OAC point called Room Average Temperature. The Diesel Generator Room temperature limit of 125°F is based on the temperature qualification of the Diesel Generator Battery Chargers. The Diesel Generator Room temperature limit of 55°F is based on the currently analyzed ambient area temperature for various diesel generator systems. The low temperature limit applies only when the Diesel Generator is not running to support standby readiness. The OPERABILITY evaluation associated with the restoration of the Diesel Generator to OPERABLE status will consider the effect of the off-limit temperature on the associated Diesel Generator SSC's.

Miscellaneous Terminal Cabinets TB208 (TB1208) and TB209 (TB1209) contain circuits associated with the function of closing the main feedwater control valves CF17, 20, 23 and 32 and main feedwater bypass valves CF104, 105, 106 and 107 on safety injection, Hi Hi S/G level, and Lo Tavg coincident with Rx trip. These cabinets also include relays used to forward the signal for main feedwater pump trip on turbine trip. TB208 is A train and TB209 is B train.

Miscellaneous Terminal Cabinets TB208 and TB1208 also contain the AMSAC inputs from main feedwater control valves CF17, 20, 23 and 32 limit switches.

Miscellaneous Terminal Cabinets TB496 (TB1496) contain circuits for IASV5080 which closes on phase A isolation to separate the air reservoir for the upper Containment Air lock from the VI system.

The maximum temperatures allowed to prevent failure of relays and fuses located in cabinets TB208, 209, 496, 1208, 1209, and 1496 are based upon manufacturer's recommendations and environmental qualification summary data.

Each air handling fan coil unit that is located in a Containment Spray (NS) pump rooms and Residual Heat Removal (ND) pump rooms provide an essential support function to the operability of the associated NS and ND pump motor. Should one of these air handling units become degraded, the operability of the affected train of the NS and ND system shall be evaluated per Technical Specification requirements in addition to requirements of SLC 16.9.16.

Although the Spent Fuel Pool Cooling (KF) pumps and motors are not contained within a Technical Specification, or subject to Operability, their function is vitally important to preventing boiling of the Spent Fuel pool. As such KF pump room temperature problems should be expeditiously resolved or alternate pool cooling would need to be implemented as committed in UFSAR section 9.1.3.

REFERENCES

1. MCC-1211.00-00-0004, Diesel Generator Ventilation Calculation.
2. MCC-1381.05-00-0313, ND Pump Motor Upper Thrust Bearing Environmental Qualification.
3. MCC-1381.05-00-0316, KF Motor Stator Thermal Life Environmental Qualification.
4. MCC-1381.05-00-0333, NS Pump Motor Lower Bearing and Oil Environmental Qualification.
5. MCC- 1240.03-00-0001, McGuire Plant Environmental Parameters (PEP) Manual
6. MCC-1223.24-00-0065, ND, NS, and KF Pump Motor Cooler Operability Evaluation
7. MCTC-1579-VD.S001-01, Diesel Generator Room Air Temperature.
8. PIPs M-93-0004, M-94-0013, M-00-1248, M-02-0015, M-04-0074, and M-03-1309.

16.9 AUXILIARY SYSTEMS

16.9.17 Refueling Operations - Decay Time

COMMITMENT The reactor shall be subcritical for at least 72 hours.

APPLICABILITY During movement of irradiated fuel in the reactor vessel.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor is subcritical for < 72 hours.	A.1 Suspend all operations involving movement of irradiated fuel in reactor vessel.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.17.1 Verify reactor has been subcritical for ≥ 72 hours by verification of date and time of subcriticality.	Prior to movement of irradiated fuel in reactor vessel

BASES

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

This SLC is limited to irradiated fuel movement within the reactor vessel. Movement of irradiated fuel assemblies from containment to the spent fuel pool is controlled through the Total Core Offloading process. A decay heat calculation may need to be performed to determine when irradiated fuel can be moved to the spent fuel pool following subcriticality.

REFERENCES

1. MCC 1227.00-00-0097, AST Dose Analysis of Fuel Handling Accidents.

2. AST Tech Spec Amendment Nos. 236/218 dated December 22, 2006.

16.9 AUXILIARY SYSTEMS

16.9.18 Refueling Operations - Communications

COMMITMENT Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY During CORE ALTERATIONS.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Direct communications between control room and refueling station personnel cannot be maintained.	A.1 Suspend CORE ALTERATIONS.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.18.1 Demonstrate direct communications between control room and personnel at refueling station.	Within 1 hour prior to start of CORE ALTERATIONS <u>AND</u> Once per 12 hours thereafter

BASES

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

REFERENCES

None

16.9 AUXILIARY SYSTEMS

16.9.19 Refueling Operations – Manipulator Crane

COMMITMENT The reactor building manipulator crane and an auxiliary hoist shall be used for movement of fuel assemblies or control rods and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1. A minimum capacity of 3250 pounds, and

   2. An overload cutoff limit $\leq$ 2900 pounds.

b. Auxiliary hoists used for latching, unlatching and drag load testing of control rods having:

   1. A minimum capacity of 1000 pounds, and

   2. A load indicator used to prevent applying a lifting force in excess of 600 pounds on the core internals.

APPLICABILITY During movement of fuel assemblies and control rods within the reactor vessel.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building crane or auxiliary hoist inoperable.	A.1 Suspend use of inoperable reactor building crane or auxiliary hoist from operations involving movement of fuel assemblies and control rods within the reactor vessel.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.19.1 Perform load test of ≥ 3250 pounds and demonstrate automatic load cutoff at ≤ 2900 pounds on each manipulator crane used for movement of fuel assemblies within the reactor vessel.	Within 30 days prior to the start of movement of fuel assemblies within the reactor vessel
TR 16.9.19.2 Perform a load test of ≥1000 pounds on each auxiliary hoist and associated load indicator used for movement of control rods or control rod drag load testing within the reactor vessel.	Within 30 days prior to the start of movement of control rods or control rod drag load testing

BASES

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

REFERENCES

None

16.9 AUXILIARY SYSTEMS

16.9.20 Crane Travel – Spent Fuel Storage Pool Building

COMMITMENT The following requirements shall be met:

a. Loads in excess of 3000 pounds shall be prohibited from travel over fuel assemblies in the storage pool,

b. Spent fuel casks shall be carried along the path outlined in Figure 16.9.20-1 in the fuel pit and fuel pool area, and

c. The requirements of LCO 3.8.2 shall be met whenever loads are moved over the spent fuel storage pool.

---NOTE---
Spent fuel pool weir gates may be moved over the stored fuel provided the decay time is $\geq$ 17.5 days since last being part of a core at power.

APPLICABILITY With fuel assemblies in the storage pool.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Place the crane load in a safe condition and suspend movement of loads over the spent fuel pool.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.20.1 Verify weight of each load, other than a fuel assembly and control rod, is < 3000 pounds.	Prior to moving the load over fuel assemblies

NOTE: Cask may move as shown in the east or west directions inside the Cask Pit once the cask has completely cleared the Cask Pit north wall.

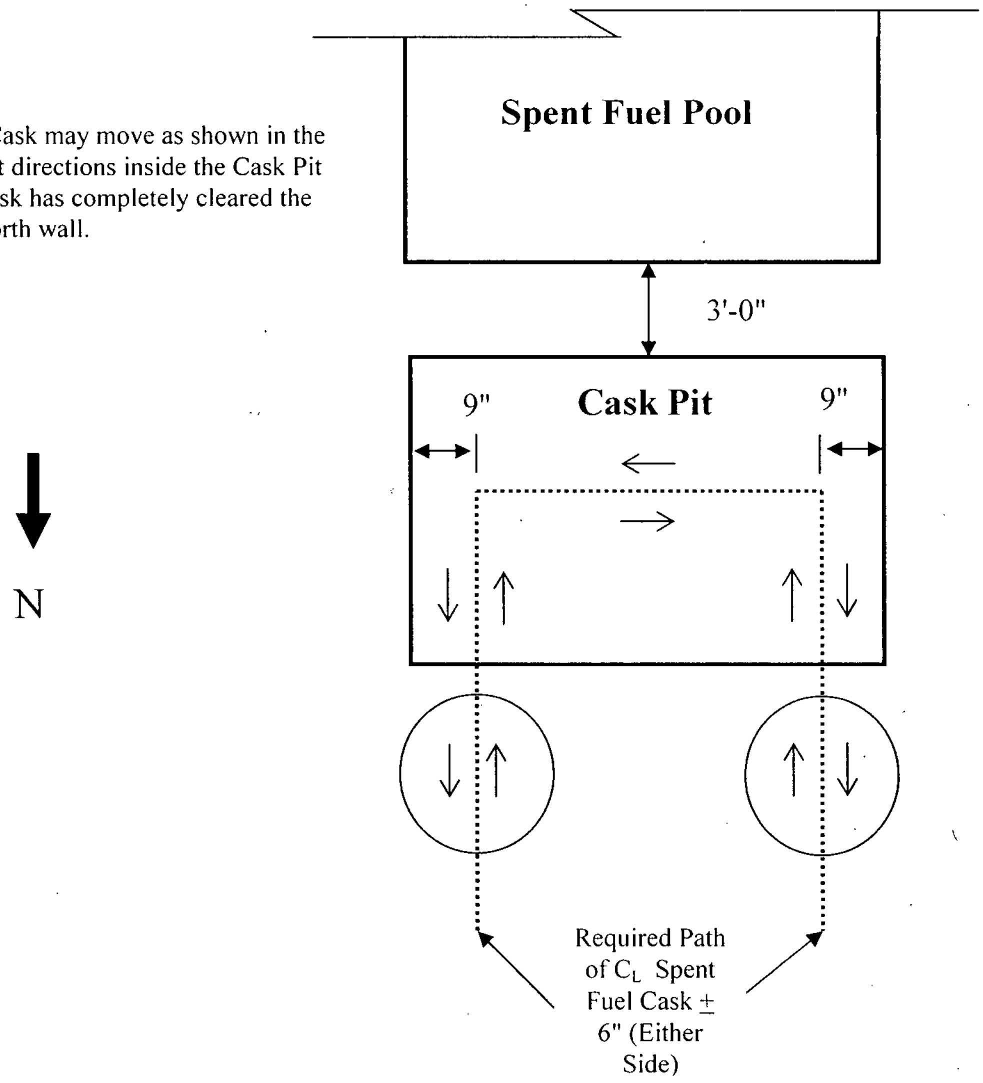


Figure 16.9.20-1

REQUIRED PATH FOR MOVEMENT OF SPENT FUEL CASKS

BASES

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analysis. The requirement for following the load path shown in Figure 16.9.20-1 assumes that the cask can not fall into the spent fuel pool.

REFERENCES

None

16.9 AUXILIARY SYSTEMS

16.9.21 Water Level – Spent Fuel Storage Pool

COMMITMENT At least 23 ft of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY Whenever irradiated fuel assemblies are being stored in the storage pool.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 Suspend all movement of fuel assemblies and crane operations with loads in the fuel storage area.	Immediately
	AND	
	A.2 Restore spent fuel storage pool water level to within limit.	4 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.21.1 Verify the water level in the spent fuel storage pool is $\geq$ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.	7 days

BASES

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

REFERENCES

None

16.9 AUXILIARY SYSTEMS

16.9.22 Switchgear Room Ventilation System (SGRVS)

COMMITMENT

1. Two trains of SGRVS for each switchgear room shall be FUNCTIONAL.
2. Temperature in each switchgear room shall be $\leq 90\,^{\circ}F$.

APPLICABILITY

Whenever the specified equipment in the switchgear room is required to be OPERABLE.

REMEDIAL ACTIONS

---NOTE---
Separate condition entry is allowed for each switchgear room.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One train of SGRVS non-functional.	A.1 Restore SGRVS to FUNCTIONAL status.	30 days
B. Two trains of SGRVS non-functional.	B.1 Restore one SGRVS to FUNCTIONAL status.	7 days
	<u>AND</u>	
	B.2 Verify switchgear room temperature is $\leq 90^{\circ}F$.	Once per 2 hours
C. Switchgear room temperature > $90^{\circ}F$.	C.1 Restore temperature to $\leq 90^{\circ}F$.	4 hours
	<u>AND</u>	
	C.2 Initiate an engineering evaluation to determine the effect of the off-limit temperature on the affected equipment.	Immediately

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9.22.1	Verify one train of SGRVS FUNCTIONAL and in service.	12 hours
TR 16.9.22.2	Verify each switchgear room is $\leq$ 90°F.	12 hours

BASES

The SGRVS (see Figure 16.9.22-1) provides cooling for the four essential switchgear rooms 1ETA, 1ETB, 2ETA, and 2ETB. Each switchgear room SGRVS has two redundant trains that provide cooling through shared ductwork. Each train consists of an air handling unit (AHU) and isolation dampers. Each AHU contains a pre-filter, water cooling coils and a fan. Air conditioning is provided through circulation of chilled water in the AHU water cooling coils. Temperature control for each switchgear room is affected by a thermostat sensing the return air of each AHU and controlling a 3-way chilled water control valve. The air is cooled to a desired temperature by exchanging heat with the chilled water. The design basis for the SGRVS is to maintain the environment in the switchgear room within an acceptable limit for the operation of unit controls.

Each train of the SGRVS is capable of maintaining the temperature in the switchgear room to less than or equal to 90°F. This temperature is limited by the Agastat timers in panels 1ATC23, 1ATC24, 2ATC23, and 2ATC24 located in the switchgear rooms. This temperature limit is provided to assure that the equipment in the room will have an acceptable service life; therefore, it will not affect the switchgear OPERABILITY. When the room temperature limit is exceeded, alternate cooling method can be used to return the temperature to within limit within 4 hours.

If both trains of SGRVS are secured, switchgear room temperature shall be verified less than or equal to 90°F once per two hours.

REFERENCES

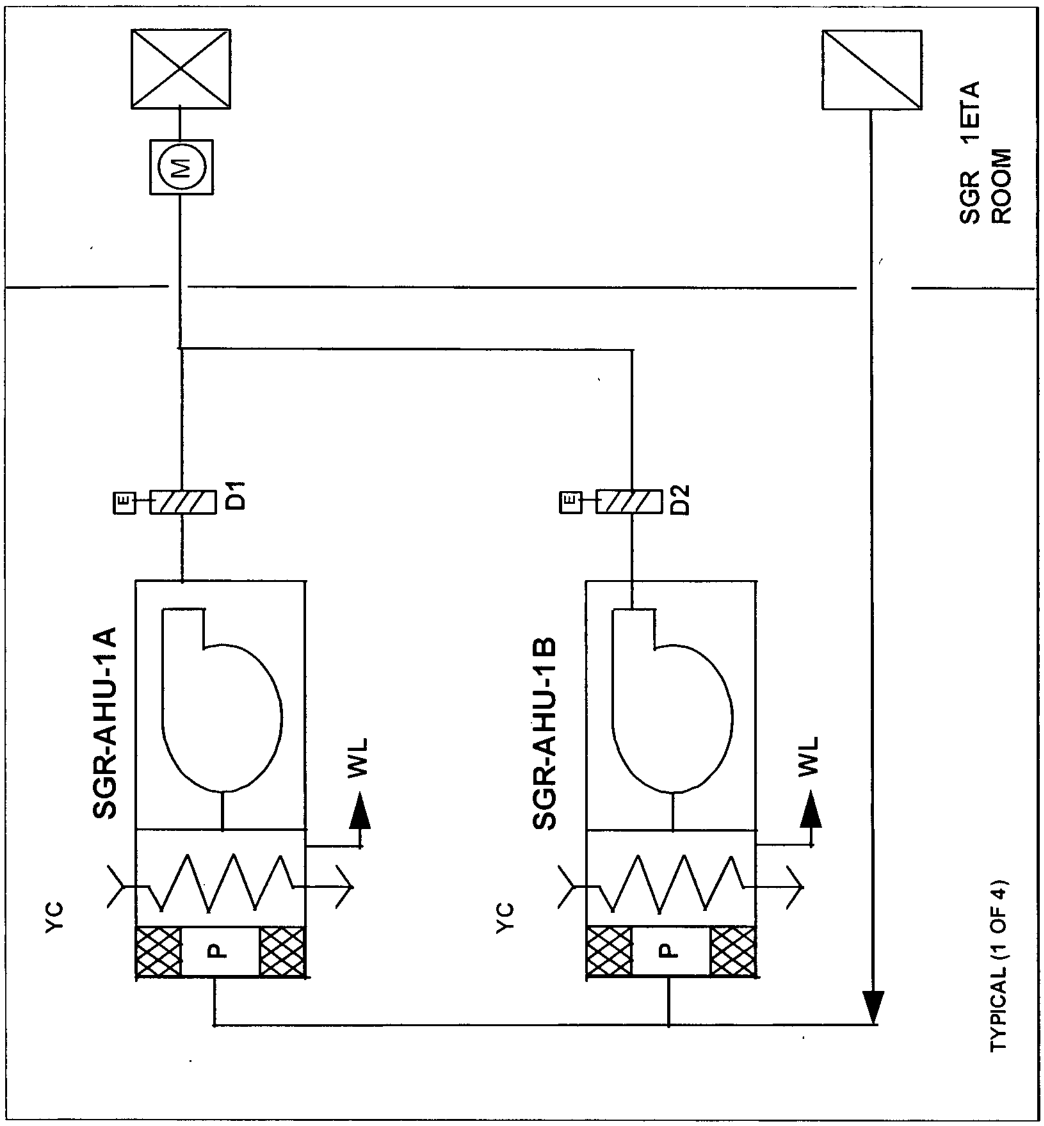
1. PIR 0-M91-0114, PIP M99-1819, PIP M99-4473, PIP M00-1604, PIP M05-5880

2. Letter dated 11/20/91, S.C. Shealy to R.R. Weidler

3. UFSAR 6.4

4. UFSAR 7.6.10

Figure 16.9.22-1 SWITCHGEAR AHUs



16.9 AUXILIARY SYSTEMS

16.9.23 Control Room Area Ventilation System (CRAVS)

COMMITMENT

1. Two trains of CRAVS shall be OPERABLE.

2. Temperature in areas listed in Table 16.9.23-1 and Table 16.9.23-2 shall be ≤ specified limits.

APPLICABILITY

Whenever the specified equipment in an affected area is required to be OPERABLE.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room area air handling unit (CRA-AHU) inoperable.	A.1 Restore CRA-AHU to OPERABLE status.	30 days
B. Two CRA-AHUs inoperable.	B.1 Restore one CRA-AHU to OPERABLE status. AND	7 days
	B.2 Verify temperature in areas listed in Table 16.9.23-1 is ≤ specified limits.	Once per 2 hours
C. One battery room exhaust fan (BR-XF) inoperable.	C.1 Restore BR-XF to OPERABLE status.	30 days

(Continued)

REMEDIAL ACTIONS (Continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two BR-XFs inoperable.	D.1	Lock open all BR-XF check dampers.	Immediately
	AND		
	D.2	Restore at least one BR-XF to OPERABLE status with all check dampers unlocked/restored.	7 days
	AND		
	D.3.1	Verify temperature in Battery Room No. 701 is $\leq 95^{\circ}F$.	Once per 2 hours
		OR	
	D.3.2	Verify temperature in each battery room listed in Table 16.9.23-2 is $\leq 104^{\circ}F$.	Once per 2 hours
	AND		
	D.4.1	Verify total flow through battery rooms is ≥ 770 cfm.	Once per 2 hours
		OR	
	D.4.2	Verify hydrogen concentration in each battery room listed in Table 16.9.23-2 is $\leq 2\%$.	Once per 2 hours

(Continued)

REMEDIAL ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Room temperature in Table 16.9.23-1 or Table 16.9.23-2, except temperature in Battery Room No. 701, not met.	E.1 Restore temperature to within limit. AND E.2 Initiate an engineering evaluation to determine the effect of the off-limit temperature on the affected equipment.	4 hours Immediately
F. Temperature in Battery Room No. 701 not met.	F.1 Verify temperature in each battery room listed in Table 16.9.23-2 is $\leq$ 104°F.	Once per 2 hours
G. Total flow through battery rooms or hydrogen concentration in battery room not met.	G.1 Restore total flow or hydrogen concentration to within limit. AND G.2 Suspend all battery equalize charging.	4 hours Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.9.23.1 Verify one CRA-AHU and one BR-XF OPERABLE and in service.	12 hours

BASES

The CRAVS (see Figure 16.9.23-1) provides cooling for the electrical penetration rooms, battery rooms, motor control center (MCC) rooms, cable rooms, restricted instrument shop, instrument storage room, and mechanical equipment room. The restricted instrument shop and instrument storage room temperature limits are not required by this SLC since they do not contain equipment vital to the operation of the plant. The CRAVS has two redundant trains. Each train consists of a non-safety control room area outside air fan (CRA-OAF), an air handling unit (AHU) and a battery room exhaust fan (BR-XF). The CRA-OAFs are not required to maintain operability of the CRAVS trains. An AHU of the CRAVS contains a pre-filter, water cooling coils and a fan. Air conditioning is provided through circulation of chilled water in the AHU water cooling coils. Temperature control for the CRAVS is affected by a room thermostat located in each of 24 zones. The air is cooled to a desired temperature by exchanging heat with the chilled water. The supply air to the battery rooms is returned via BR-XFs. These fans assist cooling by exhausting more air than is supplied. These fans also prevent hydrogen concentration from increasing to a 2% volume limit. Total flow of at least 770 cfm through the battery rooms is adequate to maintain hydrogen concentration less than or equal to 2% volume based on minimum supply flow to maintain temperature of 104°F. Hydrogen is emitted during discharging, float operation and especially during equalize charging of the batteries.

Each train of the CRAVS is capable of maintaining the temperature in the rooms to less than or equal to 104°F. This temperature is based on the maximum allowable temperature for continuous duty rating for equipment and instrumentation found in the areas served by CRAVS. The 95°F limit for Battery Room 701 in Table 16.9.23-1 is an administrative limit while the 104°F limit for Battery Rooms 706 through 711 in Table 16.9.23-2 is the actual equipment duty rating limit. This temperature limit is provided to assure that the equipment in the room will have an acceptable service life; therefore, it will not affect the battery or MCC OPERABILITY. When the room temperature limit or hydrogen concentration limit is exceeded, alternate cooling method or hydrogen purging method can be used to return the temperature or hydrogen concentration to within limit within 4 hours.

The Batteries capacities can also be affected at a minimum temperature of 60 degrees. Technical Specification Surveillance Requirement 3.8.6.3 verifies the average electrolyte temperature remains equal to or above 60 degrees. In addition, it has been shown by calculation MCC-1211.00-00-00042 that the CRAVS cannot drive the Battery room's temperature below 60 degrees. Therefore, there is no need for minimum temperature requirement within this SLC

If both CRA-AHUs or both BR-XFs are secured, verify temperature in Table 16.9.23-1 is less than or equal to the specified limits once per two hours.

BASES (Continued)

Table 16.9.23-1

ROOM NO.	DESCRIPTION	ROOM ELEV. (ft.)	MAX. TEMP. (°F)
926	ELECTRICAL PENETRATION	767	104
928	ELECTRICAL PENETRATION	767	104
808	MCCs 1EMXA AND 1EMXA-1	750	104
722	MCCs 1EMXB, 1EMXB-1, 1EMXB-2 & 1EMXB-3	733	104
821	MCCs 2EMXA, 2EMXA-1 & 1EMXH	750	104
724	MCCs 2EMXB, 2EMXB-1, 2EMXB-2 & 2EMXB-3	733	104
723A	MCC 2EMXH	733	104
801	CABLE	750	104
801C	CABLE	750	104
933	MECHANICAL EQUIPMENT	767	104
701	BATTERY	733	95

Table 16.9.23-2

ROOM NO.	DESCRIPTION	ROOM ELEV. (ft.)	MAX. TEMP. (°F)
707	BATTERY EVCA	733	104
708	BATTERY EVCB	733	104
710	BATTERY EVCC	733	104
711	BATTERY EVCD	733	104
706	BATTERY CXA	733	104
709	BATTERY CXB	733	104

REFERENCES

1. PIR 0-M91-0114, PIP M99-1819, PIP M99-4473, PIP M00-1604, PIP M-04-3742

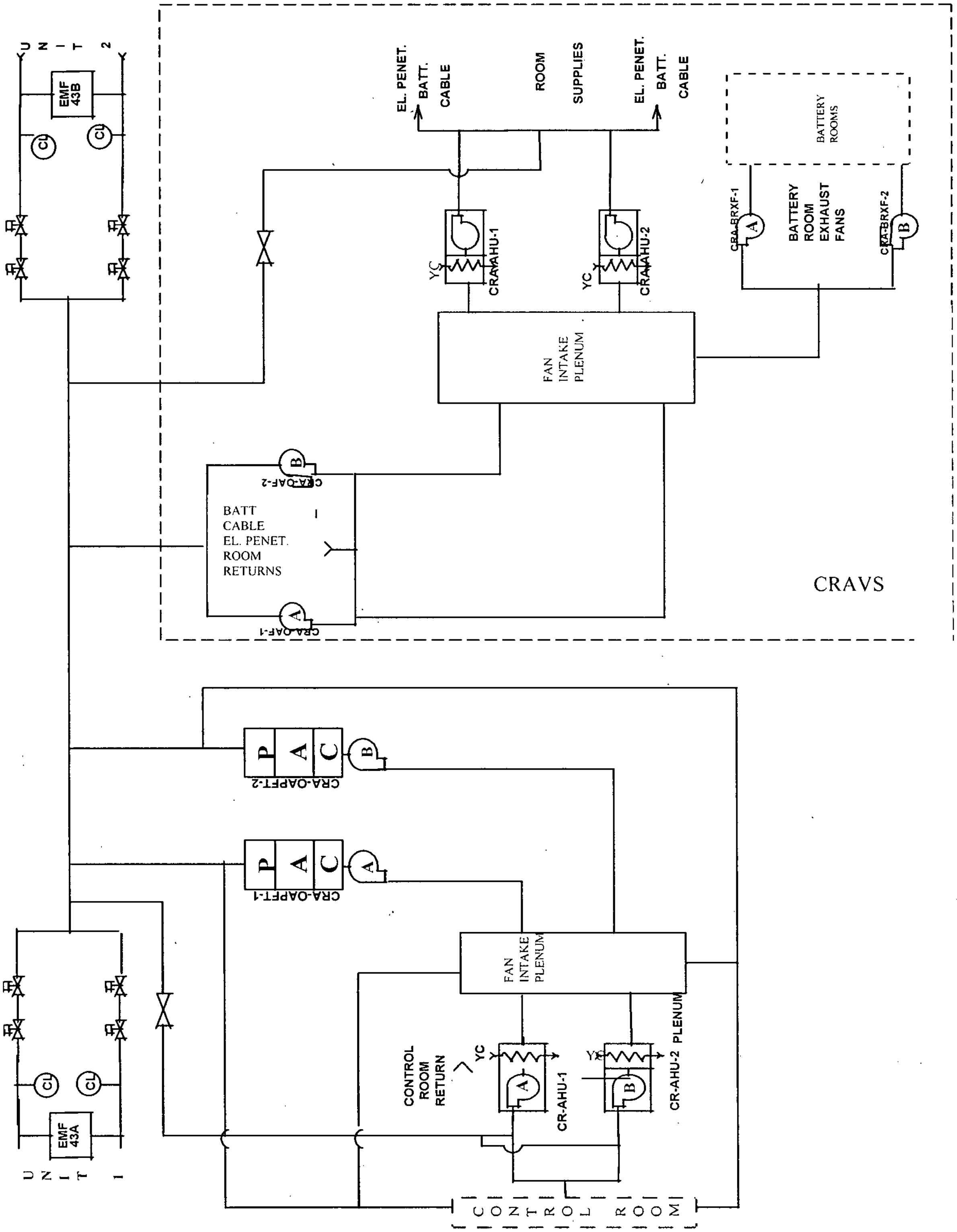
2. UFSAR 6.4

3. UFSAR 7.6.10

4. MCC-1211.00-00-00042

5. MCS-1578.VC-00-0001

Figure 16.9.23-1 **SIMPLIFIED CONTROL AREA VENTILATION SYSTEM**



16.9 AUXILIARY SYSTEMS

16.9.24 Not Used

16.9 AUXILIARY SYSTEMS

16.9.25 Refueling Operations - Containment Equipment Hatch

COMMITMENT The containment equipment hatch shall be closed and held in place by a minimum of four bolts during movement of non-recently irradiated fuel assemblies within containment.

APPLICABILITY During movement of non-recently irradiated fuel assemblies within containment.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Suspend movement of non-recently irradiated fuel assemblies within containment.	Immediately

TESTING REQUIREMENTS

None

BASES

The Selective Alternative Source Term (AST) Technical Specification (TS) License Amendment allowed relaxation of containment closure requirements during movement of non-recently irradiated fuel assemblies by revising the Applicability requirements of TS 3.9.4, Containment Penetrations. Non-recently irradiated fuel is defined as fuel that has not occupied a critical reactor core within the last 72 hours.

The re-analysis of the Fuel Handling Accidents (FHAs) using AST methodology determined that Control Room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) doses remained within regulatory limits without containment closure. After a FHA, it is advisable to close containment to further limit doses and prevent an uncontrolled airborne release. But, due to the present design of the containment equipment hatch, the hatch cannot be closed in a timely fashion without exposing workers to significant doses. All other smaller containment openings including the personnel air locks can be closed safely.

Therefore, until a safe and efficient means of closing or blocking the opening of the containment equipment hatch is developed, the containment equipment hatch shall

BASES (Continued)

remain closed during movement of non-recently irradiated fuel assemblies within containment. Good engineering practice dictates that the bolts required by this SLC Commitment be approximately equally spaced.

The control of the movement of heavy loads within containment to preclude a FHA is provided by station procedures as specified by NUREG-0612..

REFERENCES

1. AST License Amendment Request dated December 20, 2005

2. AST Tech Spec Amendment Nos. 236/218 and NRC Safety Evaluation dated December 22, 2006

3. MCC 1227.00-00-0097, AST Dose Analysis of the Fuel Handling and Weir Gate Drop Accidents

4. NSD 403 Rev. 16, Shutdown Risk Management

16.10 STEAM AND POWER CONVERSION

16.10.1 Steam Generator Pressure/Temperature Limitation

COMMITMENT Temperatures of both reactor and secondary coolants in the steam generators shall be maintained in accordance with Table 16.10.1-1.

NOTE: If steam generator level is < 10% WR, the secondary coolant temperature limit is not applicable.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Reduce steam generator pressure of the applicable side to within specified limits. <u>AND</u> A.2 Perform an engineering evaluation to determine the effect of overpressurization on the structural integrity of the steam generator and determine the steam generator remains acceptable for continued operation.	30 minutes Prior to increasing SG pressure above the specified limits.

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.10.1.1 --------------------------------NOTE-------------------------------- Only required to be performed when the temperature of either the reactor or secondary coolant is < 70 °F. -- Verify the pressure in each side of the steam generator is less than the specified limits.	1 hour

Table 16.10.1-1

TEMPERATURE	PRESSURE LIMIT
Reactor and secondary coolant temperature >70° F	No limitations by this SLC.
Lowest reactor or secondary coolant temperature ≥ 55 and ≤ 70° F	Primary pressure ≤ 400 psig. Secondary pressure ≤ 200 psig.
Lowest reactor or secondary coolant temperature < 55° F	Primary pressure ≤ 200 psig. Secondary pressure ≤ 200 psig.

BASES

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The steam generator P/T limits based on a steam generator RT_{NDT} of 0 °F and are sufficient to prevent brittle fracture.

When steam generator WR level is less than 10%, the secondary coolant temperature indications are not valid. Due to close thermal coupling of temperatures at the tube sheet, primary system temperature should be used.

REFERENCES

PIP M02-1502
MCC-1223.03-00-0049
MGMM-14512 and MGMM-14516

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.1 Liquid Effluents – Concentration

COMMITMENT The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 16.11.1-1) shall be limited:

a. For radionuclides other than dissolved or entrained noble gases, 10 times the effluent concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2, and

b. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS not within limits.	A.1 Restore the concentration to within limits.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.1.1 -------------------------------NOTE------------------------------ The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits. --- Sample and analyze radioactive liquid wastes according to Table 16.11.1-1.	According to Table 16.11.1-1

TABLE 16.11.1-1
(Page 1 of 3)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) microCi/ml (1)
1. Batch Waste Release Tanks (WMT and RMT)(4)	P Each Batch	P Each Batch	Principal Gamma Emitters(6)	5x10^{-7}
			I-131	1x10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)(7)	1x10^{-5}
	P Each Batch	M Composite(2)	H-3	1x10^{-5}
			Gross Alpha	1x10^{-7}
	P Each Batch	Q Composite(2)	Sr-89, Sr-90	5x10^{-8}
2. Continuous Releases (VUCDT discharge, CWWTS outlet and Turbine Building Sump to RC)(5)	Continuous(3)	W Composite(3)	Principal Gamma Emitters (6)	5x10^{-7}
			I-131	1x10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma emitters)(7)	1x10^{-5}
	Continuous(3)	M Composite(3)	H-3	1x10^{-5}
			Gross Alpha	1x10^{-7}
	Continuous(3)	Q Composite(3)	Sr-89, Sr-90	5x10^{-8}

TABLE 16.11.1-1
(Page 2 of 3)

NOTES:

(1) The LLD is defined, for purposes of these commitments, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\, s_b}{E \cdot V \cdot 2.22\, x\, 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microCurie per unit mass or volume),

S_b 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),

2.22×10^6 is the number of disintegrations per minute per microCurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

Typical values of E, V, Y and Δt shall be used in the calculation.

It should be recognized that the LLD is defined as an a <u>priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a</u> <u>posteriori</u> (after the fact) limit for a particular measurement.

(2) 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.

TABLE 16.11.1-1
(Page 3 of 3)

(3) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously or intermittently in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

(4) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and thoroughly mixed to assure representative sampling.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

(6) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. The LLD for Ce-144 is 5×10^{-6} microCi/ml. 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 be identified and reported in the Annual Radioactive Effluent Release Report.

(7) The principal gas gamma emitters for which the LLD specification applies are Xe-133 and Xe-135. These are the reference nuclides in Regulatory Guide 1.21.

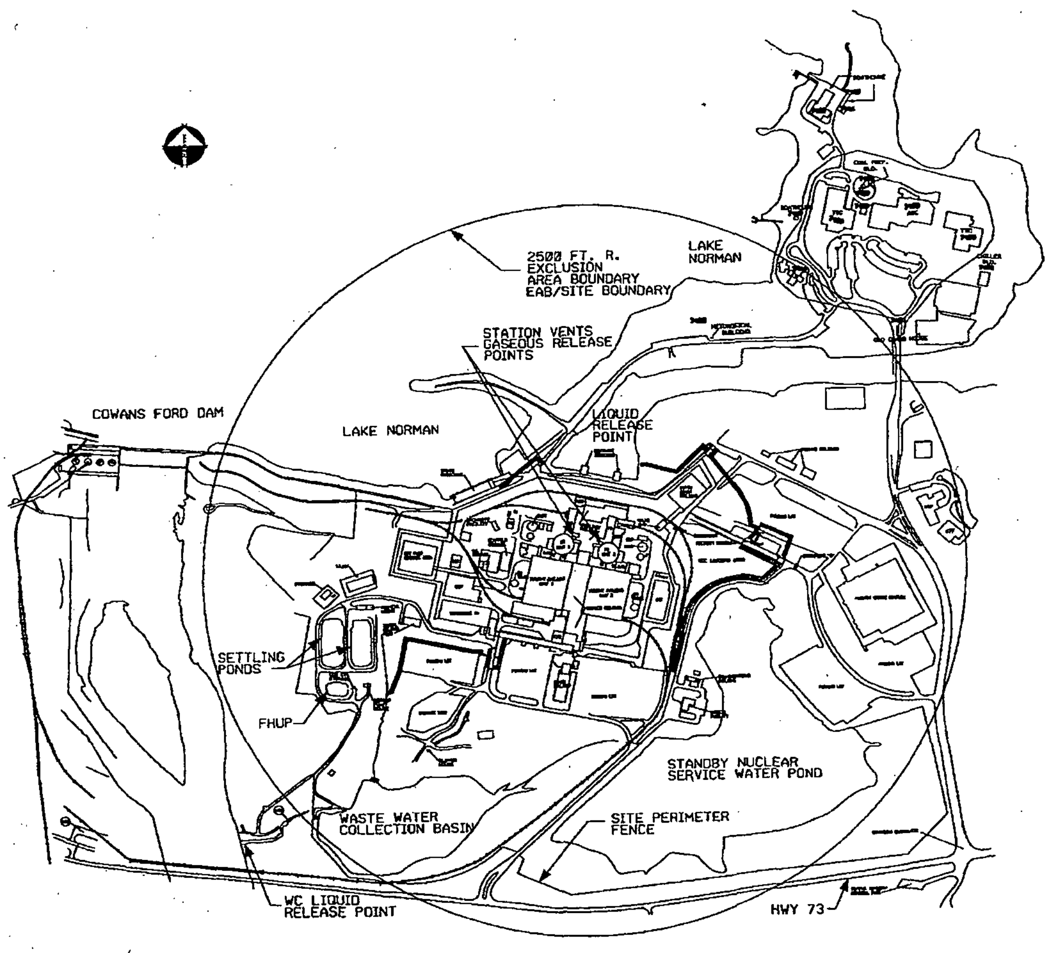


FIGURE 16.11.1-1 SITE BOUNDARY / EXCLUSION AREA BOUNDARY

BASES

This commitment is provided to ensure that the concentration of radioactive materials released in liquid waste effluents 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. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.1301 to the population. The concentration limit for dissolved or entrained 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. This commitment applies to the release of liquid effluents from all reactors at the site.

The basic requirements for the Selected Licensee Commitments concerning effluents from nuclear power reactors are stated in 10CFR50.36a. These requirements indicate that compliance with effluent Selected Licensee Commitments will keep average annual releases of radioactive material in effluents to small percentages of the limits specified in the old 10CFR20.106 (new 10CFR20.1301). These requirements 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. 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 as is reasonably achievable (ALARA) as set forth in 10CFR50, Appendix I.

As stated in the Introduction to Appendix B of the new 10CFR20, the effluent concentration (EC) limits given in Appendix B, Table 2, Column 2, are based on an annual dose of 50 mrem. Since a release concentration corresponding to a limiting dose rate of 500 mrem/year has been acceptable as a SLC limit for liquid effluents, which applies at all times as an assurance that the limits of 10CFR50, Appendix I are not likely to be exceeded, it should not be necessary to reduce this limit by a factor of 10.

Operational history at Catawba/McGuire/Oconee has demonstrated that the use of the concentration values associated with the old 10CFR20.106 as SLC limits has resulted in calculated maximum individual doses to members of the public that are small percentages of the limits of 10CFR50, Appendix I. Therefore, the use of concentration values which correspond to an annual dose of 500 mrem should not have a negative impact on the ability to continue to operate within the limits of 10CFR50 Appendix I and 40CFR190.

Having sufficient operational flexibility is especially important in establishing a basis for effluent monitor setpoint calculations. As discussed above, the concentrations stated in the new 10CFR20, Appendix B, Table 2, Column 2, relate to a dose of 50 mrem in a year. When applied on an instantaneous basis, this corresponds to a dose rate of 50 mrem/year. This low value is impractical upon which to base effluent monitor setpoint calculations for many liquid effluent release situations when monitor background, monitor sensitivity, and monitor performance must be taken into account.

BASES (continued)

Therefore, to accommodate operational flexibility needed for effluent releases, the limits associated with SLC 16.11.1 are based on ten times the concentrations stated in the new 10CFR20, Appendix B, Table 2, Column 2 to apply at all times. The multiplier of ten is proposed because the annual dose of 500 mrem, upon which the concentrations in the old 10CFR20, Appendix B, Table II, Column 2 are based, is a factor of ten higher than the annual dose of 50 mrem, upon which the concentrations in the new 10CFR20, Appendix B, Table 2, Column 2, are based. Compliance with the limits of the new 10CFR20.1301 will be demonstrated by operating within the limits of 10CFR50, Appendix I and 40CFR190.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

REFERENCES

1. McGuire Nuclear Station Offsite Dose Calculation Manual (ODCM)

2. International Commission on Radiological Protection (ICRP) Publication 2

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.2 Radioactive Liquid Effluent Monitoring Instrumentation

COMMITMENT The radioactive liquid effluent monitoring instrumentation channels shown in Table 16.11.2-1 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of SLC 16.11.1 are not exceeded.

AND

The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY As shown in Table 16.11.2-1.

REMEDIAL ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more radioactive liquid effluent monitoring channels Alarm/Trip setpoint less conservative than required.	A.1 Suspend the release of radioactive liquid effluents monitored by the affected channel. OR A.2 Declare the channel inoperable. OR A.3 Adjust setpoint to within limit.	Immediately Immediately Immediately
B. One or more radioactive liquid effluent monitoring instrument channels inoperable.	B.1 Enter the Remedial Action specified in Table 16.11.2-1 for the channel(s).	Immediately

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel inoperable.	C.1.1 Analyze two independent samples per TR 16.11.1.1.	Prior to initiating a release
	AND	
	C.1.2 Perform independent verification of the discharge line valving.	Prior to initiating a release
	AND	
	C.1.3.1 Perform independent verification of manual portion of the computer input for the release rate calculations performed by computer.	Prior to initiating a release
	OR	
	C.1.3.2 Perform independent verification of entire release rate calculations for calculations performed manually.	Prior to initiating a release
	AND	
	C.1.4 Restore channel to OPERABLE status.	14 days
	OR	
	C.2 Suspend the release of radioactive effluents via this pathway.	Immediately

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more channels inoperable.	D.1 Obtain grab samples from the effluent pathway.	Once per 12 hours during releases.
	AND	
	D.2 Perform an analysis of grab samples for radioactivity.	To meet LLD requirements per Table 16.11.1-1.
	AND	
	D.3 Restore the channel to OPERABLE status.	30 days
E. One or more flow rate measurement channels inoperable.	E.1 -------------NOTE--------------- Pump performance curves generated in place may be used to estimate flow. ----------------------------------- Estimate the flow rate of the release.	Once per 4 hours during releases
	AND	
	E.2 Restore the channel to OPERABLE status.	30 days
F. RC minimum flow interlock inoperable.	F.1 Verify that the number of pumps providing dilution is greater than or equal to the number of pumps required.	Once per 4 hours during releases
	AND	
	F.2 Restore the channel to OPERABLE status.	30 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition C, D, E or F not met.	G.1 Explain why the inoperability was not corrected within the specified Completion Time in the Annual Radioactive Effluent Release Report.	In the next scheduled Annual Radioactive Effluent Release Report

TESTING REQUIREMENTS

---NOTE---
Refer to Table 16.11.2-1 to determine which TRs apply for each Radioactive Liquid Effluent Monitoring channel.

TEST		FREQUENCY
TR 16.11.2.1	Perform CHANNEL CHECK.	24 hours
TR 16.11.2.2	------------------------------NOTE------------------------------ The CHANNEL CHECK shall consist of verifying indication of flow. -- Perform CHANNEL CHECK.	Every 24 hours during periods of release
TR 16.11.2.3	Perform SOURCE CHECK.	Prior to each release
TR 16.11.2.4	Perform SOURCE CHECK.	31 days
TR 16.11.2.5	------------------------------NOTES------------------------------ 1. For Instrument 1, the COT shall also demonstrate that automatic isolation of the pathway occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint. 2. For Instruments 1 and 2, the COT shall also demonstrate that control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint; circuit failure and, a downscale failure. -- Perform CHANNEL OPERATIONAL TEST.	92 days
TR 16.11.2.6	Perform a CHANNEL CALIBRATION.	18 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.11.2.7	---------------------------NOTE--------------------------- The initial CHANNEL CALIBRATION shall be performed using standards certified by the National Institute of Standards and Technology (NIST) or using standards obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. -- Perform a CHANNEL CALIBRATION.	24 months

TABLE 16.11.2-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	REMEDIAL ACTION	APPLICABILITY	TESTING REQUIREMENTS
1. Radioactivity Monitors Providing Alarm And Automatic Termination of Release				
a. Waste Liquid Effluent Line (EMF-49)	1 per station	A, C, G	During liquid effluent releases	TR 16.11.2.1 TR 16.11.2.3 TR 16.11.2.5 TR 16.11.2.7
b. EMF-49 Minimum Flow Device (2)	1 per station	C, G	During liquid effluent releases	TR 16.11.2.5 TR 16.11.2.7
c. Containment Ventilation Unit Condensate Line (EMF-44)	1	A, D, G	At all times	TR 16.11.2.1 TR 16.11.2.4 TR 16.11.2.5 TR 16.11.2.7
d. EMF-44 Minimum Flow Device (2)	1	D, G	At all times	TR 16.11.2.5 TR 16.11.2.7
2. Radioactivity Monitors Providing Alarm But Not Automatic Termination of Release				
a. Conventional Waste Water Treatment Line or Turbine Building Sump to RC (EMF-31)	1	A, D, G	At all times	TR 16.11.2.1 TR 16.11.2.4 TR 16.11.2.5 TR 16.11.2.7
b. EMF-31 Minimum Flow Device (2)	1	D, G	At all times	TR 16.11.2.5 TR 16.11.2.7
3. Continuous Composite Samplers				
a. Containment Ventilation Unit Condensate Line	1	D, G	At all times	TR 16.11.2.2 TR 16.11.2.5 TR 16.11.2.6
b. Conventional Waste Water Treatment Line	1 per station	D, G	At all times	TR 16.11.2.2 TR 16.11.2.5 TR 16.11.2.6
c. Turbine Building Sump to RC	1	D, G	At all times	TR 16.11.2.2 TR 16.11.2.6

(Continued)

4. Flow Rate Measurement Devices				
a. Waste Liquid Effluent Line	1 per station	E, G	During liquid effluent releases	TR 16.11.2.2 TR 16.11.2.5 TR 16.11.2.6
b. Containment Ventilation Unit Condensate Line	1	E, G	At all times	TR 16.11.2.2 TR 16.11.2.5 TR 16.11.2.6
c. Conventional Waste Water Treatment Line	1 per station	E, G	At all times	TR 16.11.2.2 TR 16.11.2.5 TR 16.11.2.6
d. Turbine Building Sump to RC	1	E, G	At all times	TR 16.11.2.2 TR 16.11.2.6
5. RC Minimum Flow Interlock (1)	1 per station	F, G	At all times	TR 16.11.2.5

NOTES:

1. Minimum flow dilution is assured by an interlock which terminates waste liquid release if the number of RC pumps running falls below the number of pumps required for dilution. The required number of RC pumps for dilution is determined per station procedures.

2. Radioactivity Monitor (EMF) shall not be declared operable unless both the EMF and the associated EMF's Minimum Flow Device are rendered operable.

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 of liquid effluents. The minimum flow devices for EMFs listed in Table 16.11.2-1 are required to provide assurance of representative sampling during actual or potential releases of liquid effluents. An interlock between the EMF's minimum flow device and its associated flow rate measurement device disables the remove alarm during non-release timeframes for the purpose of the control room black board annunciator criteria that disable expected alarms. An EMF flow rate measurement device measures total flow of the effluent while the EMF minimum flow device measures the sample flow rate through the EMF. The Alarm/Trip Setpoints of these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the Alarm/Trip will occur prior to exceeding the limits stated in SLC 16.11.1. 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 Turbine Building Sump to RC Discharge Flow Measurement and Sampler Devices are for monitoring only and do not alarm or have any controls that require a COT.

REFERENCES

1. McGuire Nuclear Station Offsite Dose Calculation Manual (ODCM)

2. 10 CFR Part 50, Appendix A

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.3 Dose - Liquid Effluents

COMMITMENT

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS (see Figure16.11.1-1) shall be limited:

a. During any calendar quarter, to ≤ 1.5 mrem to the total body and to ≤ 5 mrem to any organ, and

b. During any calendar year, to ≤ 3 mrem to the total body and to ≤ 10 mrem to any organ.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

---NOTES---

Enter applicable Conditions and Required Actions of SLC 16.11.12, "Total Dose," when the limits of this SLC are exceeded by twice the specified limit.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose from release of radioactive materials in liquid effluents exceeding above limits.	---------------------NOTE-------------- The Special Report shall include the 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, Safe Drinking Water Act, as applicable. --	
	A.1 Prepare and submit a Special Report to the NRC which identifies the causes for exceeding the limits, corrective actions taken to reduce releases, and actions taken to ensure that subsequent releases are within limits.	30 days

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.3.1 Determine cumulative dose contributions from liquid effluents for current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM.	31 days

BASES

This commitment is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The commitment implements the guides set forth in Section II.A of Appendix I. The REMEDIAL ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that 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 Part 141. These requirements are applicable only if the drinking water supply is taken from the river 3 miles downstream of the plant discharge.

The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation 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 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This commitment applies to the release of liquid effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system in accordance with the guidance given in NUREG-0133, Chapter 3.1.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 40 CFR Part 141, Safe Drinking Water Act

3. 10 CFR Part 50, Appendix I

4. Regulatory Guide 1.109, "Calculation 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 1, October 1977.

5. Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.4 Liquid Radwaste Treatment System

COMMITMENT

The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent from each unit to UNRESTRICTED AREAS (see Figure16.11.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radioactive liquid waste being discharged without treatment and in excess of above limits. AND Any portion of Liquid Radwaste Treatment System not in operation.	A.1 Prepare and submit a Special Report to the NRC which identifies the reasons liquid radwaste was discharged without treatment, identification of inoperable equipment and reasons for inoperability, corrective actions taken to restore the equipment to OPERABLE status, and actions taken to prevent recurrence.	30 days

TESTING REQUIREMENTS

---NOTE---
The Liquid Radwaste Treatment System shall be demonstrated OPERABLE by meeting SLC 16.11.1 and16.11.3.

TEST		FREQUENCY
TR 16.11.4.1	Project liquid release doses from each unit to UNRESTRICTED AREAS, in accordance with the methodology and parameters in the ODCM, when water systems are being released without being processed by its radwaste treatment system.	31 days

BASES

The requirement 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 specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

This commitment applies to the release of liquid effluents from each reactor at the site. For units with shared Radwaste Treatment Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system in accordance with the guidance given in NUREG-0133, Chapter 3.1.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR Part 50

3. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.5 Chemical Treatment Ponds

COMMITMENT

The quantity of radioactive material contained in each chemical treatment pond shall be limited by the following expression (excluding tritium and dissolved or entrained noble gases):

$$\frac{264}{V} \cdot \sum_{j} \frac{A_j}{(C_j \; x \, 10)} < 1.0$$

Where:

A_j = pond inventory limit for single radionuclide "j", in Curies

C_j = 10 CFR 20, Appendix B, Table 2, Column 2, concentration for single radionuclide "j", microCuries/ml;

V = design volume of liquid and slurry in the pond, in gallons; and

264 = conversion unit, microCuries/Curie per milliliter/gallon.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Quantity of radioactive material in any of the chemical treatment ponds exceeding above limit.	A.1 Suspend all additions of radioactive material to the pond.	Immediately
	<u>AND</u>	
	A.2 Initiate corrective action to reduce the pond contents to within limits.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.5.1 Verify quantity of radioactive material in each batch of slurry (powdex resin) to be transferred to chemical treatment ponds is within limits by analyzing a representative sample of the slurry. Each batch to be transferred to the chemical treatment ponds is limited by: $$\sum_j \frac{Q_j}{(C_j \, x \, 10)} < 6.0 \, x \, 10^5 \, \frac{pCi/gm}{\mu Ci/ml}$$	Prior to each transfer

BASES

The inventory limits of the chemical treatment ponds (CTP) are based on limiting the consequences of an uncontrolled release of the pond inventory. The expression in SLC 16.11.5 assumes the pond inventory is uniformly mixed, that the pond is located in an uncontrolled area as defined in 10 CFR Part 20, and that the concentration limit in Note 4 to Appendix B of 10 CFR Part 20 applies.

The batch limits of slurry to the chemical treatment ponds assure that radioactive material in the slurry transferred to the CTP are "as low as is reasonably achievable" in accordance with 10 CFR Part 50.36a. The expression in SLC 16.11.5 assures no batch of slurry will be transferred to the CTP unless the sum-of the ratios of the activity of the radionuclides to their respective concentration limitation is less than the ratio of the 10 CFR Part 50, Appendix I, Section II.A, total body dose level to the instantaneous whole body dose rate limitation, or that:

$$\sum_j \frac{c_j}{(C_j \, x \, 10)} < \frac{3 \, mrem/yr}{500 \, mrem/yr} = 0.006$$

Where:

c_j = Radioactive slurry concentration for radionuclide "j" entering the UNRESTRICTED AREA chemical treatment ponds, in microCuries/milliliter; and

C_j = 10 CFR 20, Appendix B, Table 2, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

BASES (continued)

For the design of filter/demineralizers using powder resin, the slurry wash volume and the weight of resin used per batch is fixed by the cell surface area, and the slurry volume to resin weight ratio is constant at 100 ml/gram of wet, drained resin with a moisture content of approximately 55 to 60% (bulk density of about 58 pounds per cubic feet). Therefore,

$$\sum_j \frac{c_j}{(C_j \, x \, 10)} = \sum_j \frac{Q_j}{(C_j \, x \, 10)\,(10^2 \; ml/gm)\,(10^6 \; pCi/\mu Ci)} < 0.006, \text{ and}$$

$$\sum_j \frac{Q_j}{(C_j \, x \, 10)} < 6.0 \, x \, 10^5 \; \frac{pCi/gm}{\mu Ci/ml}$$

Where:

Q_j = concentration of radioactive materials in wet, drained slurry (powdex resin) for radionuclide "j", excluding tritium, dissolved or entrained noble gases, and radionuclides with less than an 8-day half-life. The analysis shall include at least Ce-144, Cs-134, Cs-137, Co-58 and Co-60, in picoCuries/gram. Estimates of the Sr-89 and Sr-90 batch concentration shall be included based on the most recent monthly composite analysis (within 3 months); and

C_j = 10 CFR 20, Appendix B, Table 2, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

The batch limits provide assurance that activity input to the chemical treatment ponds will be minimized, and a means of identifying radioactive material in the inventory limitation of SLC 16.11.5.

The basic requirements for the Selected Licensee Commitments concerning effluents from nuclear power reactors are stated in 10CFR50.36a. These requirements indicate that compliance with effluent Selected Licensee Commitments will keep average annual releases of radioactive material in effluents to small percentages of the limits specified in the old 10CFR20.106 (new 10CFR20.1301). These requirements 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. 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 as is reasonably achievable (ALARA) as set forth in 10CFR50, Appendix I.

BASES (continued)

As stated in the Introduction to Appendix B of the new 10CFR20, the effluent concentration (EC) limits given in Appendix B, Table 2, Column 2, are based on an annual dose of 50 mrem. Since a release concentration corresponding to a limiting dose rate of 500 mrem/year has been acceptable as a SLC limit for liquid effluents, which applies at all times as an assurance that the limits of 10CFR50, Appendix I are not likely to be exceeded, it should not be necessary to reduce this limit by a factor of 10.

Operational history at Catawba/McGuire/Oconee has demonstrated that the use of the concentration values associated with the old 10CFR20.106 as SLC limits has resulted in calculated maximum individual doses to members of the public that are small percentages of the limits of 10CFR50, Appendix I. Therefore, the use of concentration values which correspond to an annual dose of 500 mrem should not have a negative impact on the ability to continue to operate within the limits of 10CFR50, Appendix I and 40CFR190.

Having sufficient operational flexibility is especially important in establishing a basis for effluent monitor setpoint calculations. As discussed above, the concentrations stated in the new 10CFR20, Appendix B, Table 2, Column 2, relate to a dose of 50 mrem in a year. When applied on an instantaneous basis, this corresponds to a dose rate of 50 mrem/year. This low value is impractical upon which to base effluent monitor setpoint calculations for many liquid effluent release situations when monitor background, monitor sensitivity, and monitor performance must be taken into account.

Therefore, to accommodate operational flexibility needed for effluent releases, the limits associated with SLC 16.11.1 are based on ten times the concentrations stated in the new 10CFR20, Appendix B, Table 2, Column 2 to apply at all times. The multiplier of ten is proposed because the annual dose of 500 mrem, upon which the concentrations in the old 10CFR20, Appendix B, Table II, Column 2 are based, is a factor of ten higher than the annual dose of 50 mrem, upon which the concentrations in the new 10CFR20, Appendix B, Table 2, Column 2, are based. Compliance with the limits of the new 10CFR20.1301 will be demonstrated by operating within the limits of 10CFR50, Appendix I and 40CFR190.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR 20, Appendix B

3. 10 CFR 50, Appendix I, Section II.A

4. 10 CFR 20

5. 10 CFR 50.36a

16.11 RADIOLOGICAL EFFLUENT CONTROL

16.11.6 Dose Rate - Gaseous Effluents

COMMITMENT

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure16.11.1-1) shall be limited to the following:

a. For noble gases: ≤ 500 mrem/yr to the whole body and ≤ 3000 mrem/yr to the skin, and

b. For Iodine - 131 and 133, for tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days: ≤ 1500 mrem/yr to any organ.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Dose rate not within limit.	A.1 Restore the release rate to within limits.	Immediately

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.6.1 Verify dose rates due to noble gases in gaseous effluents are within limits in accordance with the methodology and parameters in the ODCM.	In accordance with the ODCM
TR 16.11.6.2 Verify dose rates due to radioactive materials, other than noble gases, in gaseous effluents are within limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with Table16.11.6-1.	In accordance with Table 16.11.6-1

TABLE 16.11.6-1
(Page 1 of 4)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[1] (μCi/ml)
1. Waste Gas Storage Tanks	P Each Tank Grab Sample	P Each Tank	Principal Gas Gamma Emitters[6]	1×10^{-4}
2. Containment Purge	P Each PURGE Grab Sample	P Each PURGE	Principal Gas Gamma Emitters[6]	1×10^{-4}
		M	H-3	1×10^{-6}
3. Unit Vent	W[2] Grab Sample	W	Principal Gas Gamma Emitters[6]	1×10^{-4}
			H-3	1×10^{-6}
4.a. Radwaste Facility Vent b. Waste Handling Building c. Equipment Staging Building	W Grab Sample	W	Principal Gas Gamma Emitters[6]	1×10^{-4}
			H-3	1×10^{-6}
5. Unit Vents	Continuous[5]	W[8] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[5]	W[8] Particulate Sample	Principal Gamma Emitters[6] (I-131, Others)	1×10^{-11}
	Continuous[5]	M Composite Particulate Sample	Gross Alpha[7]	1×10^{-11}
	Continuous[5]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

TABLE 16.11.6-1
(Page 2 of 4)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[(1)] (μCi/ml)
6. All Release Types as listed in 4 above.	Continuous[(5)]	W[(8)] Charcoal Sample	I-131	$1x10^{-12}$
			I-133	$1x10^{-10}$
	Continuous[(5)]	W[(8)] Particulate Sample	Principal Gamma Emitters[(6)] (I-131, Others)	$1x10^{-11}$
	Continuous[(5)]	M Composite Particulate Sample	Gross Alpha[(7)]	$1x10^{-11}$
	Continuous[(5)]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$

TABLE 16.11.6-1
(Page 3 of 4)

NOTES:

1. The LLD is defined, for purposes of these commitments, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

   For a particular measurement system (which may include radiochemical separation):

$$\text{LLD} = \frac{4.66\, s_b}{E \cdot V \cdot 2.22\, x\, 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

   Where:

LLD	=	the "a priori" lower limit of detection as defined above (as microCurie per unit mass or volume);
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute);
E	=	the counting efficiency (as counts per disintegration);
V	=	the sample size (in units of mass or volume);
2.22×10^6	=	the number of disintegrations per minute per microCurie;
Y	=	the fractional radiochemical yield (when applicable);
λ	=	the radioactive decay constant for the particular radionuclide; and
Δt	=	the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

   Typical values of E, V, Y and Δt shall be used in the calculation.

   It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

TABLE 16.11.6-1
(Page 4 of 4)

NOTES:

2. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

3. Not used.

4. Not used.

5. The ratio of the sample flow volume to the sampled stream flow volume shall be known for the time period covered by each dose or dose rate calculation made in accordance with SLCs 16.11.6, 16.11.8 and 16.11.9.

6. The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, and Ce-141 in iodine and particulate releases. The LLD for Ce-144 is $5x10^{-9}$ microCi/ml. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report.

7. The composite filter(s) will be analyzed for alpha activity by analyzing the filter media used during the collection period.

8. Samples shall be changed at least once per 7 days and analyses shall be completed to meet LLD after changing, or after removal from sampler. If the particulate and charcoal sample frequency is changed to a 24 hour frequency the corresponding LLDs may be increased by a factor of 10 (i.e., LLD for I-131 from 1×10^{-12} to 1×10^{-11} microCi/ml).

BASES

Specific release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the whole body, and 3000 mrem/year to the skin from noble gases, and 1500 mrem/year to any organ from Iodine 131, Iodine 133, tritium, and all radionuclides in particulate form with half-lives greater than eight days. This commitment applies to the release of gaseous effluents from all reactors at the site. The Exclusion Area Boundary (Site Boundary) is set as the boundary for gaseous effluent release limits. The Exclusion Area Boundary (EAB) is formed by a 2500 ft radius centered on the Reactor Buildings' centerlines as shown on Figure 16.11.1-1.

The basic requirements for the Selected Licensee Commitments concerning effluents from nuclear power reactors are stated in 10CFR50.36a. These requirements indicate that compliance with effluent Selected Licensee Commitments will keep average annual releases of radioactive material in effluents to small percentages of the limits specified in the old 10CFR20.106 (new 10CFR20.1301). These requirements 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. 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 as is reasonably achievable (ALARA) as set forth in 10CFR50, Appendix I.

As stated in the Introduction to Appendix B of the new 10CFR20, the effluent concentration (EC) limits given in Appendix B, Table 2, Column 1, are based on an annual dose of 50 mrem for isotopes for which inhalation or ingestion is limiting or 100 mrem for isotopes for which submersion (noble gases) is limiting. Since release concentrations corresponding to limiting dose rates of less than or equal to 500 mrem/year to the whole body, 3000 mrem/year to the skin from noble gases, and 1500 mrem/year to any organ from Iodine 131, Iodine 133, tritium and for all radionuclides in particulate form with half-lives greater than eight days at the site boundary has been acceptable as a SLC limit for gaseous effluents to assure that the limits of 10CFR50, Appendix I and 40CFR190 are not likely to be exceeded, it should not be necessary to restrict the operational flexibility by incorporating the EC value for isotopes based on ingestion/inhalation (50 mrem/year) or for isotopes with the EC based on submersion (100 mrem/year).

Having sufficient operational flexibility is especially important in establishing a basis for effluent monitor setpoint calculations. As discussed above, the concentrations stated in the new 10CFR20, Appendix B, Table 2, Column 1, relate to a dose of 50 or 100 mrem in a year. When applied on an instantaneous basis, this corresponds to a dose rate of either 50 or 100 mrem/year. These low values are impractical upon which to base effluent monitor setpoint calculations for many effluent release situations when monitor background, monitor sensitivity, and monitor performance must be taken into account. Therefore, to accommodate operational flexibility needed for effluent releases, the limits associated with SLC 16.11.6 will be maintained at the current dose rate limit for noble gases of 500 mrem/year to the whole body and 3000 mrem/year to the skin, for Iodine 131, Iodine 133, tritium and all radionuclides in particulate form with half-lives greater than eight days an instantaneous dose rate limit of 1500 mrem/year to any organ.

BASES (continued)

Compliance with the limits of the new 10CFR20.1301 will be demonstrated by operating within the limits of 10CFR50, Appendix I and 40CFR190. Operational history at Catawba/McGuire/Oconee has demonstrated that the use of the dose rate values listed above (i.e. 500 mrem/year, 3000 mrem/year and 1500 mrem/year) as SLC limits has resulted in calculated maximum individual doses to members of the public that are small percentages of the limits of 10CFR50, Appendix I and 40CFR190.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K. "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual
2. 10 CFR Part 20, Appendix B
3. 10 CFR Part 20
4. 10 CFR Part 50

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.7 Radioactive Gaseous Effluent Monitoring Instrumentation

COMMITMENT

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 16.11.7-1 shall be OPERABLE with Alarm/Trip Setpoints set to ensure that the limits of SLC 16.11.6 are not exceeded.

AND

The Alarm/Trip setpoints shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

-----NOTE-----
Brief periods of routine sampling (not to exceed 15 minutes) do not make the instrumentation inoperable.

APPLICABILITY

As shown in Table 16.11.7-1.

REMEDIAL ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more radioactive gaseous effluent monitoring channels Alarm/Trip setpoint less conservative than required.	A.1 Suspend the release of radioactive gaseous effluents monitored by the affected channel.	Immediately
	OR	
	A.2 Declare the channel inoperable.	Immediately
	OR	
	A.3 Adjust setpoint to within limit.	Immediately

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more radioactive gaseous effluent monitoring instrument channels inoperable.	B.1 Enter the Remedial Action specified in Table 16.11.7-1 for the channel(s).	Immediately
C. One channel inoperable.	C.1.1 Analyze two independent samples of the tank contents. AND C.1.2 Perform independent verification of the discharge valve lineup. AND C.1.3.1 Perform independent verification of manual portion of the computer input for the release rate calculations performed by computer. OR C.1.3.2 Perform independent verification of entire release rate calculations for calculations performed manually. AND C.1.4 Restore channel to OPERABLE status. OR C.2 Suspend the release of radioactive effluents via this pathway.	Prior to initiating a release Prior to initiating a release Prior to initiating a release Prior to initiating a release 14 days Immediately

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more flow rate measurement channels inoperable.	D.1 Estimate the flow rate of the release. AND D.2 Restore the channel to OPERABLE status.	Once per 4 hours during releases 30 days
E. One or more noble gas activity monitor channels inoperable.	E.1 Obtain grab samples from the effluent pathway. AND E.2 Perform an analysis of grab samples for radioactivity. AND E.3 Restore the channel to OPERABLE status.	Once per 12 hours during releases To meet LLD requirements per Table 16.11.6-1 30 days
F. Noble gas activity monitor providing automatic termination of release inoperable.	F.1 Suspend PURGING or VENTING of radioactive effluents via this pathway.	Immediately
G. One or more sampler channels inoperable.	G.1 Perform sampling with auxiliary sampling equipment as required by Table 16.11.6-1. AND G.2 Restore the channel to OPERABLE status.	Continuously 30 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. One or more Sampler Minimum Flow Device Channels inoperable.	H.1 Verify flow through the sampling apparatus.	Once per 4 hours during releases
	<u>AND</u>	
	H.2 Restore the channel to OPERABLE status.	30 days
I. Required Action and associated Completion Time of Condition C, D, E, F, G, or H not met.	I.1 Explain why the inoperability was not corrected within the specified Completion Time in the Annual Radioactive Effluent Release Report.	In the next scheduled Annual Radioactive Effluent Release Report

TESTING REQUIREMENTS

---NOTE---

Refer to Table 16.11.7-1 to determine which TRs apply for each Radioactive Gaseous Effluent Monitoring channel.

TEST	FREQUENCY
TR 16.11.7.1 Perform CHANNEL CHECK.	Prior to each release
TR 16.11.7.2 ------------------------------NOTE------------------------------ The SOURCE CHECK for these channels shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity or a simulated source of radioactivity such as a light emitting diode. --- Perform SOURCE CHECK.	Prior to each release
TR 16.11.7.3 Perform CHANNEL CHECK.	24 hours
TR 16.11.7.4 Perform CHANNEL CHECK.	7 days

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.11.7.5	-------------------------------NOTE--------------------------------- The SOURCE CHECK for these channels shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity or a simulated source of radioactivity such as a light emitting diode. -- Perform SOURCE CHECK.	31 days
TR 16.11.7.6	--------------------------------NOTES------------------------------- 1. For noble gas activity monitors providing automatic termination of release, the COT shall also demonstrate that automatic isolation of the pathway occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint. 2. For all noble gas activity monitors, the COT shall also demonstrate that control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint; circuit failure and, a downscale failure. -- Perform CHANNEL OPERATIONAL TEST.	92 days
TR 16.11.7.7	---------------------------NOTE------------------------------------- For all noble gas activity monitors, the initial CHANNEL CALIBRATION shall be performed using standards certified by the National Institute of Standards and Technology (NIST) or using standards obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. -- Perform a CHANNEL CALIBRATION.	18 months

TABLE 16.11.7-1
(Page 1 of 3)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENTS	MINIMUM CHANNELS OPERABLE	REMEDIAL ACTION	APPLICABILITY	TESTING REQUIREMENTS
1. WASTE GAS HOLDUP SYSTEM				
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Low Range-EMF-50 or 1EMF-36, low-range)	1 per station	A, C, I	During gas effluent releases.	TR 16.11.7.1 TR 16.11.7.2 TR 16.11.7.6 TR 16.11.7.7
b. Effluent System Flow Rate Measuring Device	1 per station	D, I	At all times except when isolation valve is closed & locked.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
2. Condenser Evacuation System - Noble Gas Activity Monitor (EMF-33)	1	A, E, I	When air ejectors are operable.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
3. Vent System				
a. Noble Gas Activity Monitor (Low Range - EMF-36)	1	A, E, I	At all times.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
b. Iodine Sampler	1	G, I	At all times, except during routine sampling.	TR 16.11.7.4
c. Particulate Sampler (EMF-35)	1	G, I	At all times, except during routine sampling.	TR 16.11.7.4
d. Unit Vent Flow Rate Monitor (Totalizer)	1	D, I	At all times.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
e. Iodine Sampler Minimum Flow Device	1	H,I	At all times, except during routine sampling.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
f. Particulate Sampler Minimum Flow Device (1)	1	G,I	At all times, except during routine sampling.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
4. Containment Purge System - Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Low Range - EMF-39)	1	A, F, I	Modes 1 through 6, except when isolation valve is closed & locked.	TR 16.11.7.2 TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7

(continued)

TABLE 16.11.7-1
(Page 2 of 3)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENTS	MINIMUM CHANNELS OPERABLE	REMEDIAL ACTION	APPLICABILITY	TESTING REQUIREMENTS
5. Auxiliary Building Ventilation System - Noble Gas Activity Monitor (EMF-41 or EMF-36)	1	A, E, I	At all times.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
6. Fuel Storage Area Ventilation System - Noble Gas Activity Monitor (EMF-42 or EMF-36)	1	A, E, I	At all times.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
7. Contaminated Parts Warehouse Ventilation System				
a. Noble Gas Activity Monitor (EMF-53)	1 per station	A, E, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
b. Flow Rate Monitor	1 per station	D, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
c. EMF-53 Sampler Minimum Flow Device (1)	1 per station	H,I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
8. Radwaste Facility Ventilation System				
a. Noble Gas Activity Monitor (EMF-52)	1 per station	A, E, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
b. Flow Rate Monitor	1 per station	D, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
c. EMF-52 Sampler Minimum Flow Device (1)	1 per station	H, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7

(continued)

TABLE 16.11.7-1
(Page 3 of 3)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENTS	MINIMUM CHANNELS OPERABLE	REMEDIAL ACTION	APPLICABILITY	TESTING REQUIREMENTS
9. Equipment Staging Building Ventilation System				
a. Noble Gas Activity Monitor (EMF-59)	1 per station	A, E, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7
b. Flow Rate Monitor	1 per station	D, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
c. EMF-59 Sampler Minimum Flow Device (1)	1 per station	H, I	During gaseous effluent releases.	TR 16.11.7.3 TR 16.11.7.6 TR 16.11.7.7
10. Containment Air Release and Addition System - Noble Gas Activity Monitor (EMF-39L or EMF-36L)	1	A, E, I	At all times except when isolation valve is closed & locked.	TR 16.11.7.3 TR 16.11.7.5 TR 16.11.7.6 TR 16.11.7.7

NOTES:

1. Radioactivity monitor (EMF) shall not be declared OPERABLE unless both the EMF and the associated EMF's Minimum Flow Device are rendered OPERABLE.

BASES

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 of gaseous effluents. The instrumentation consists of monitoring and sampling instrumentation. Monitors provide continuous display of process parameters with appropriate alarms and trip setpoints established. Samplers collect a portion of the desired process for subsequent laboratory analysis, and do not have alarm/trip capability. Samplers and the analysis program provide a method to assure that long term effluent release quantities do not exceed the requirements of SLC 16.11.6. Monitors provide assurance that instantaneous effluent releases do not exceed the requirements of SLC 16.11.6. The minimum flow devices for EMFs listed in Table 16.11.7-1 are required to provide assurance of representative sampling during actual or potential releases of gaseous effluents. The flow rate monitor quantifies the total gaseous effluent (both non-radioactive and radioactive) released to the environment. During routine sampling, instrumentation may be turned off for short periods of time (not to exceed 15 minutes) in order to meet analysis requirements of SLC 16.11.6. This is considered to be a normal operable function of the equipment. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the Alarm/Trip will occur prior to exceeding the limits stated in SLC 16.11.6. 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.

REFERENCES

1. McGuire Nuclear Station, Offsite Dose Calculation Manual

2. 10 CFR Part 50, Appendix A

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.8 Noble Gases

COMMITMENT

Air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure16.11.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

---NOTES---

Enter applicable Conditions and Required Actions of SLC 16.11.12, "Total Dose," when the limits of this SLC are exceeded by twice the specified limit.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated air dose from radioactive noble gases in gaseous effluents exceeding any of above limits.	A.1 Prepare and submit a Special Report to the NRC which identifies the causes for exceeding the limits, corrective actions taken to reduce releases, and actions taken to ensure that subsequent releases are within limits.	30 days

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.8.1 Determine cumulative dose contributions from noble gases in gaseous effluents for current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM.	31 days

BASES

This commitment is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I.

The REMEDIAL ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable."

The TESTING REQUIREMENTS implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially under-estimated.

The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation 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 1, 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," Revision 1, July 1977.

The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This commitment applies at all times to the release of gaseous effluents from each reactor at the site. For units with shared Radwaste Treatment Systems, the gaseous effluents from the shared system are to be proportioned among the units sharing that system in accordance with the guidance given in NUREG-0133, Chapter 3.1.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.9 Dose - Iodine-131 and 133, Tritium and Radioactive Materials in Particulate Form

COMMITMENT

The dose to a MEMBER OF THE PUBLIC from Iodine-131 and 133, tritium, and all radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas at and beyond the SITE BOUNDARY (see Figure 16.11.1-1) shall be limited to the following:

a. During any calendar quarter: less than or equal to 7.5 mrem to any organ, and

b. During any calendar year: less than or equal to 15 mrem to any organ.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

-----NOTES-----

Enter applicable Conditions and Required Actions of SLC 16.11.12, "Total Dose," when the limits of this SLC are exceeded by twice the specified limit.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose from the release of Iodine 131 and 133, tritium, and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents exceeding any of the above limits.	A.1 Prepare and submit a Special Report to the NRC which identifies the causes for exceeding the limits, corrective actions taken to reduce releases, and actions taken to ensure that subsequent releases are within limits.	30 days

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.9.1 Determine cumulative dose contributions for Iodine 131 and 133, tritium, and radioactive material in particulate form with half lives greater than 8 days in gaseous effluents for current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM.	31 days

BASES

This commitment is provided to implement the requirements-of Sections- II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I.

The REMEDIAL ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable.

The ODCM calculational methods specified in the TESTING REQUIREMENTS implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated.

The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation 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 1, 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, Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides; (2) deposition of radionuclides onto 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.

BASES (continued)

This commitment applies at all times to the release of gaseous effluents from each reactor at the site. For units with shared Radwaste Treatment Systems, the gaseous effluents from the shared system are to be proportioned among the units sharing that system in accordance with the guidance given in NUREG 0133, Chapter 3.1.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.10 Gaseous Radwaste Treatment System

COMMITMENT

The VENTILATION EXHAUST TREATMENT and WASTE GAS HOLDUP SYSTEMS shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 16.11.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radioactive gases being discharged without treatment and in excess of above limits.	A.1 Prepare and submit a Special Report to the NRC which identifies inoperable equipment and reasons for inoperability, actions taken to restore the equipment to OPERABLE status, and actions taken to prevent recurrence.	30 days

TESTING REQUIREMENTS

---NOTE---
The installed Gaseous Radwaste Treatment System shall be demonstrated OPERABLE by meeting SLC 16.11.6, 16.11.8 and16.11.9.

TEST	FREQUENCY
TR 16.11.10.1 Project gaseous release doses from each unit to areas at and beyond the SITE BOUNDARY, in accordance with the methodology and parameters in the ODCM, when gaseous systems are being released without being processed by its radwaste treatment system.	31 days

BASES

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable."

This commitment implements the requirements of 19 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This commitment applies at all times to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the gaseous effluents from the shared system are to be proportioned among the units sharing that system in accordance with NUREG-0133, Chapter 3.1.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR Part 50, Appendix I

3. 10 CFR Part 50

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.11 Solid Radioactive Waste

COMMITMENT

Radioactive wastes shall be processed and packaged to ensure compliance with the applicable requirements of 10 CFR Part 20, 10CFR Part 61, 10 CFR Part 71, and State regulations governing the transportation and disposal of radioactive wastes.

The Solid Radwaste System or an approved alternative process shall be used in accordance with a PROCESS CONTROL PROGRAM (PCP) for the solidification of liquid or wet radioactive wastes or the dewatering of wet radioactive wastes to be shipped for direct disposal at a 10CFR61 licensed disposal site. Wastes shipped for off site processing in accordance with the processor's specifications and transportation requirements are not required to be solidified or dewatered to meet disposal requirements.

- The PCP describes administrative and operational controls used for the solidification of liquid or wet solid radioactive wastes in order to meet applicable 10CFR61 waste form requirements.
- The PCP describes the administrative and operational controls used for the dewatering of wet radioactive wastes to meet 10CFR61 free standing water requirements.
- The process parameters used in establishing the PCP shall be based on demonstrated processing of actual or simulated liquid or wet solid wastes and must adequately verify that the final product of solidification or dewatering meets all applicable Federal, State and disposal site requirements.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Applicable regulatory requirements for solidified or dewatered wastes are not satisified.	A.1 Suspend shipments of defectively packaged solid radioactive wastes from the site.	Immediately
	<u>AND</u>	
	A.2 Initiate action to correct the PROCESS CONTROL PROGRAM, procedures, or solid waste equipment as necessary to prevent recurrence.	Prior to next shipment for disposal of solidified or dewatered wastes.
B. A solidification test as described in the PCP fails to verify Solidification.	B.1 Suspend solidification of the batch under test and follow PCP guidance for test failures.	Immediately
	B.2 Once a subsequent test verifies Solidification, solidification of the batch may then be resumed as directed by the PCP. The PCP shall be modified as required to assure Solidification of subsequent batches of waste	Prior to next solidification for shipment of waste for disposal at a 10CFR61 disposal site.

(continued)

REMEDIAL ACTIONS (continued)

C. With solidification or dewatering for disposal not performed in accordance with the PROCESS CONTROL PROGRAM.	C.1 Reprocess the waste in accordance with PCP requirements. OR C.2 Follow PCP or procedure quidance for alternative free standing liquid verification to ensure the waste in each container meets disposal requirements and take appropriate administrative action to prevent recurrence.	Prior to shipment for disposal of the inadequately processed waste that requires solidification of dewatering
D. With the solid waste equipment incapable of meeting SLC 16.11.11 or not in service	D.1 Restore the equipment to OPERABLE status or provide for alternative capability to process wastes as necessary to satisfy all applicable disposal requirements	In a time frame that supports the COMMITMENT section of SLC 16.11.11

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.11.1 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 radioactive waste to be solidified for disposal at a 10CFR61 disposal site per the COMMITMENT of this SLC.	Every tenth batch of each type of radioactive waste to be solidified.

BASES:

This commitment implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and requirements to use a Process Control Program to meet applicable 10CFR61 waste form criteria for solidified and dewatered radioactive wastes.

REFERENCES:

1. 10CFR Part 50, "Domistic Licensing of Production and Utilization Facilities"

2. 10 CFR Part 50, Appendix A

3. 10CFR20, "Standards for Protection Against Radiation"

4. 10CFR61, "Licensing Requirements for Land Disposal of Radioactive Waste

5. 10CFR71, "Packaging and Transportation of Radioactive Materials"

6. DPCo Process Control Program Manual

7. NRC Generic Letter 84-12, "Compliance With 10 CFR Part 61 And Implementation Of the Radiological Effulent Technical Specifications (Rets) and Attendant Process Control Program (PCP)"

8. NRC Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effulent Technical Specifications In the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of Rets to the Offsite Dose Calculation Manual or to the Process Control Program"

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.12 Total Dose

COMMITMENT The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to ≤ 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to ≤ 75 mrem.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated doses from releases exceeding twice the specified limits of SLC 16.11.3, 16.11.8 or 16.11.9.	A.1 Verify, by calculation, the cumulative dose from direct radiation contributions, the ISFSI, outside storage tanks, and radioactivity releases are within the total dose limit.	Immediately
	AND	
	A.2 --------------NOTE------------ Only required to be performed if the total dose limit is exceeded. ------------------------------------	
	Prepare and submit a Special Report to the NRC which identifies corrective actions to be taken to reduce subsequent releases to prevent recurrence and schedule for achieving conformance with specified limits.	30 days

TESTING REQUIREMENTS

--NOTE--

Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with SLC 16.11.3, 16.11.8 and16.11.9, and in accordance with the methodology and parameters specified in the ODCM.

TEST	FREQUENCY
TR 16.11.12.1 Determine cumulative dose contributions from direct radiation from the units, the ISFSI, and from radwaste storage tanks in accordance with the methodology and parameters specified in the ODCM.	When calculated doses from effluent releases exceeds twice the limits of SLCs 16.11.3, 16.11.8 or 16.11.9

BASES

This commitment is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of 10 CFR Part 50, Appendix I, and if direct radiation doses from the units and outside storage tanks are kept small.

This Special Report, as defined in 10 CFR Part 20.2203(a)(4), 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 levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER of the PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered.

If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in

BASES (continued)

accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.2203(a)(4), is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 and a variance is granted until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in SLCs 16.11.1 and 16.11.6.

An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

REFERENCES

1. McGuire Nuclear Station, Offsite Dose Calculation Manual
2. 10 CFR Part 20
3. 40 CFR Part 190
4. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT MONITORING

16.11.13 Radiological Environmental Monitoring Program

COMMITMENT The Radiological Environmental Monitoring Program shall be conducted as specified in Table 16.11.13-1.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Radiological Environmental Monitoring Program not being conducted as specified in Table 16.11.13-1.	A.1	Identify the reasons for not conducting the program as required and the plans for preventing a recurrence in the Annual Radiological Environmental Operating Report.	Within the next scheduled Annual Radiological Environmental Operating Report
B.	Radioactivity level of environmental sampling medium at a specified location in excess of reporting limits of Table 16.11.13-2.	B.1	Prepare and submit a Special Report that defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of SLC 16.11.3, 16.11.8, and 16.11.9.	30 days

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Milk or fresh leafy vegetable samples unavailable from one or more required sample locations.	C.1 --------------NOTE------------- Specific locations from which samples were unavailable may be deleted from the program. ---------------------------------- Revise the Radiological Environmental Monitoring Program to identify locations for obtaining replacement samples.	30 days
	AND	
	C.2 Identify the cause of the unavailability of samples and identify new location(s) for obtaining replacement samples in the next Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).	Within the next scheduled Annual Radioactive Effluent Release Report

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.13.1 -------------------------------NOTES------------------------------- The maximum values for the lower limits of detection shall be as specified in Table16.11.13-3. --- The radiological environmental monitoring samples shall be collected from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables16.11.13-1.	In accordance with Table 16.11.13-1

TABLE 16.11.13-1
(Page 1 of 6)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[(1)]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation[(2)]	Forty routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site; and The balance of the stations placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly	Gamma dose quarterly.

(continued)

TABLE 16.11.13-1
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RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne Radioiodine and Particulates	Samples from five locations: Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground level D/Q. One sample from the vicinity of a community having the highest calculated annual average ground level D/Q. One sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction(3).	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(4); Gamma isotopic analysis(5) of composite (by location quarterly).
3. Waterborne a. Surface(6)	One sample upstream. One sample downstream.	Composite sample over 1-month period(7).	Gamma isotope analysis(5) monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources only if likely to be affected(8)	Quarterly	Gamma isotopic(5) and tritium analysis quarterly.

(continued)

TABLE 16.11.13-1
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RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
c. Drinking	One sample of each of one to three of the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period[7] when I-131 analysis is performed; monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year[9]. Composite for gross beta and gamma isotopic analyses[5] monthly. Composite for tritium analysis quarterly.
d. Sediment from the shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually	Gamma isotopic analysis[5] semiannually.
4. Ingestion a. Milk	Samples from milking animals in three locations within 5-km distance having the highest dose potential. If there are none, then one sample from milking animals in each of three areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per year[9]. One sample from milking animals at a control location 15 to 30 km distant and in the least prevalent wind direction.	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic[5] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

(continued)

TABLE 16.11.13-1
(Page 4 of 6)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
b. Fish and Invertebrates	One sample each commercially and recreationally important species in vicinity of plant discharge area. One sample of same species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis(5) on edible portions
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest(10)	Gamma isotopic analyses(5) on edible portion.
	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly, when available.	Gamma isotopic(5) and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly, when available.	Gamma isotopic(5) and I-131 analysis.

TABLE 16.11.13-1

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RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

NOTES:

1. Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 16.11.13-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report. It is recognized that, at times, it may not be possible or practical to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program. In lieu of an Licensee Event Report, identify the cause of the unavailability of samples for that pathway and identify the new locations(s) for obtaining replacement samples in the next Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

2. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The forty stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sections will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

3. The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

4. Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 16.11.13-1

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RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

NOTES (continued):

5. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

6. The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

7. A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

8. Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

9. The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

10. If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuborous and root food products.

TABLE 16.11.13-2
(Page 1 of 1)

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	REPORTING LEVELS				
ANALYSIS	WATER (pCi/l)	AIRBOURNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	BROAD LEAF VEGETATION (pCi/kg, wet)
H-3	20,000[(1)]	N/A	N/A	N/A	N/A
Mn-54	1,000	N/A	30,000	N/A	N/A
Fe-59	400	N/A	10,000	N/A	N/A
Co-58	1,000	N/A	30,000	N/A	N/A
Co-60	300	N/A	10,000	N/A	N/A
Zn-65	300	N/A	20,000	N/A	N/A
Zr-Nb-95	400	N/A	N/A	N/A	N/A
I-131	2	0.9	N/A	3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200	N/A	N/A	300	N/A

NOTES:

1. For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

TABLE 16.11.13-3
(Page 1 of 3)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD) (1)(2)(3)

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	0.01	N/A	N/A	N/A	N/A
H-3	2000*	N/A	N/A	N/A	N/A	N/A
Mn-54	15	N/A	130	N/A	N/A	N/A
Fe-59	30	N/A	260	N/A	N/A	N/A
Co-58, 60	15	N/A	130	N/A	N/A	N/A
Zn-65	30	N/A	260	N/A	N/A	N/A
Zr-95	15	N/A	N/A	N/A	N/A	N/A
Nb-95	15	N/A	N/A	N/A	N/A	N/A
I-131	1(4)	0.07	N/A	1	60	N/A
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	15	N/A	N/A	15	N/A	N/A
La-140	15	N/A	N/A	15	N/A	N/A

* If no drinking water pathway exists, a value of 3000 pCi/l may be used.

TABLE 16.11.13-3
(Page 2 of 3)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)

NOTES:

1. The LLD is defined, for purposes of these commitments, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will 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):

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

   Where:

   LLD is the "a priori" lower limit of detection as defined above (as picoCurie per unit mass or volume),

   s_b 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, and

   Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

   It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an 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, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

TABLE 16.11.13-3
(Page 3 of 3)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)

NOTES (continued):

2. This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report.

3. Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

4. LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

BASES

The Radiological Environmental Monitoring Program is established to monitor the radiation and radionuclides in the environs of the plant. The program provides representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program is contained in SLC 16.11.13 – 16.11.16 and conforms to the guidance of Appendix I to 10 CFR Part 50. The program includes 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.

The portion of the Radiological Environmental Monitoring Program required by this commitment provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and 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 the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 16.11.13-3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

With the level of radioactivity in an environmental sampling medium at a specified location exceeding the reporting levels of Table 16.11.13-3 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days a Special Report that defines the corrective actions to be

BASES (continued)

taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of SLCs 16.11.6, 16.11.8, and 16.11.9. When more than one of the radionuclides in Table 16.11.13-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots\ldots \geq 1.0$$

When radionuclides other than those in Table 16.11.13-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of SLCs 16.11.6, 16.11.8 and 16.11.9. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report. The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.14 Land Use Census

COMMITMENT

A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of:

a. the nearest milk animal,

b. the nearest residence, and

c. the nearest garden of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall identify within a distance of 5 km (3 miles) the location in each of the 16 meteorological sectors of:

a. all milk animals, and

b. all gardens of greater than 50 m^2 producing broad leaf vegetation.

---NOTE---
Broad leaf vegetation sampling of three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 16.11.13-1 4c shall be followed, including analysis of control samples.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Location(s) identified which yields a calculated dose/dose commitment greater than values currently calculated in SLC 16.11.9.	A.1 Identify the new location in the Annual Radioactive Effluent Release Report.	In next scheduled Annual Radioactive Effluent Release Report

(continued)

REMEDIAL ACTIONS (continued)

B. Location(s) identified which yields a calculated dose or dose commitment (via same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with SLC 16.11.13.	B.1	Add the new location to the Radiological Environmental Monitoring Program.	30 days
	AND		
	B.2	--------------NOTES----------- If samples cannot be obtained, an explanation of why samples are not obtainable (substitute representative locations if possible) shall be included. --------------------------------- Identify the new location(s), revised figures and tables for the ODCM, in the next Annual Radiological Release Report.	In the next scheduled Annual Radiological Release Report

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.14.1 ----------------------------NOTE-------------------------------- The results of the land use census shall be included in the Annual Radiological Environmental Operating Report. --- Conduct a land use census during the growing season using the information which will provide the best results such as a door-to-door survey, aerial survey, or consultation with local agricultural authorities.	12 months

BASES

This commitment is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with SLC 16.11.13, add the new location to the Radiological Environmental Monitoring Program. The sampling location(s), excluding the control station 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.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.15 Interlaboratory Comparison Program

COMMITMENT Analyses shall be performed on radioactive materials, supplied as part of an Interlaboratory Comparison Program (ICP), that correspond to samples required by SLC 16.11.13.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Analyses not being performed as required.	A.1 Report corrective actions taken to prevent recurrence in the Annual Radiological Environmental Operating Report.	In next scheduled Annual Radiological Environmental Operating Report

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.15.1 Report a summary of the results of the Interlaboratory Comparison Program in the Annual Radiological Environmental Operating Report.	12 months

BASES

This requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

The Interlaboratory Comparison Program (ICP) shall be described in the Annual Radiological Environmental Operating Report.

REFERENCES

1. 10 CFR Part 50, Appendix I

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.16 Annual Radiological Environmental Operating Report

COMMITMENT

Routine Annual 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 Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with pre-operational studies, with operational controls as appropriate, and with 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 land use censuses required by SLC 16.11.14.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual 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 possible in a supplementary report.

The reports shall also include the following:

- a summary description of the Radiological Environmental Monitoring Program;
- at least two legible maps covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor (one map shall cover stations near the site boundary; a second shall include the more distant stations);
- the results of licensee participation in the Interlaboratory Comparison Program, required by SLC 16.11.15;
- a discussion of all deviations from the sampling schedule of Table 16.11.13-1; and

COMMITMENT (continued)

- a discussion of all analyses in which the LLD required by Table 16.11.13-3 was not achievable.

A single submittal may be made for a multiple unit station..

APPLICABILITY

At all times.

REMEDIAL ACTIONS

None

TESTING REQUIREMENTS

None

BASES

None

REFERENCES

1. Technical Specification 5.6.2

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.17 Radioactive Effluent Release Reports

COMMITMENT

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report shall include an annual summary of hourly meteorological data collected over the previous calendar year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. A five year average of representative onsite meteorological data shall be used in the gaseous effluent dose pathway calculations. Dispersion factors (X/Qs) and deposition factors (D/Qs) shall be generated using the computer code XOQDOQ (NUREG/CR-2919) which implements NRC Regulatory Guide 1.111. The meteorological conditions concurrent with the time of release shall be reviewed annually to determine if the five-year average values should be revised. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The 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 Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

COMMITMENT (continued)

The Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite or disposed of in the site landfill during the report period:

a. Total container volume, in 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., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Number of shipments, and

f. Solidification agent or absorbent (e.g., cement, or other approved agents (media)).

The Radioactive Effluent Release Reports 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 Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and 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 pursuant to SLC 16.11.14.

The Radioactive Effluent Release Reports shall also identify any licensee initiated major changes to the Radioactive Waste Systems (liquid, gaseous, and solid). Otherwise, this information may be included in the annual UFSAR update. The discussion of each change shall contain:

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

COMMITMENT (continued)

e. An evaluation of the change, which shows expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the Station Manager or the Chemistry Manager.

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

APPLICABILITY

At all times

REMEDIAL ACTIONS

None

TESTING REQUIREMENTS

None

BASES

None

REFERENCES

1. Technical Specification 5.6.3

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.18 Liquid Holdup Tanks

COMMITMENT — The quantity of radioactive material contained in each unprotected outdoor radwaste tank shall be limited to ≤ 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY — At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Quantity of radioactive material in tank not within limit.	A.1 Suspend all additions of radioactive material to the tank.	Immediately
	<u>AND</u>	
	A.2 Reduce the tank contents to within limit.	48 hours
	<u>AND</u>	
	A.3 Describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.	Within the next scheduled Annual Radioactive Effluent Release Report

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.18.1 Verify the quantity of radioactive material contained in unprotected outdoor radwaste tanks is within limits by analyzing a representative sample of the tank's contents when radioactive materials are being added to the tank.	7 days

BASES

The tanks applicable to this SLC include all those outdoor radwaste 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 quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR 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

None

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.19 Explosive Gas Mixture

COMMITMENT The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to ≤ 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of oxygen in the WASTE GAS HOLDUP SYSTEM > 2% but ≤ 4% by volume.	A.1 Reduce oxygen concentration to within limits.	48 hours
B. Concentration of oxygen in the WASTE GAS HOLDUP SYSTEM > 4% and hydrogen concentration > 4% by volume.	B.1 Suspend all additions of waste gases to the system.	Immediately
	<u>AND</u>	
	B.2 Reduce the concentration of oxygen to ≤ 4% by volume.	Immediately
	<u>AND</u>	
	B.3 Reduce oxygen concentration to within limits.	48 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.19.1 Verify the concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM is within limits by monitoring waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required by SLC 16.7.8.	During WASTE GAS HOLDUP SYSTEM operation

BASES

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

REFERENCES

None

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.20 Gas Storage Tanks

COMMITMENT The quantity of radioactivity contained in each gas storage tank shall be limited $\leq$ 49,000 Curies noble gases (considered as Xe-133).

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Quantity of radioactive material in tank not within limit.	A.1 Suspend all additions of radioactive material to the tank.	Immediately
	AND	
	A.2 Reduce the tank contents to within limit.	48 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.20.1 Verify the quantity of radioactive material contained in each gas storage tank is within limit when radioactive materials are being added to the tank.	24 hours

BASES

This SLC considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the dose guideline values of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

REFERENCES

None

16.12 RADIATION PROTECTION

16.12.1 In-Plant Iodine Monitoring

COMMITMENT

A program shall be established, implemented, and maintained which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

APPLICABILITY

At all times.

REMEDIAL ACTIONS

None

TESTING REQUIREMENTS

None

BASES

This commitment is provided to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions.

REFERENCES

1. UFSAR 11.4, Process and Effluent Radiological Monitoring Systems, January 1, 1998.

2. Technical Specification 6.8.4.b, as amended through amendments 166/148.

3. Technical Specification 3.3.3.6 as amended through amendments 166/148.

4. NUREG-0737, III.D.3.3.

16.12 RADIATION PROTECTION

16.12.2 Sealed Source Contamination

COMMITMENT Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Sealed source having removable contamination in excess of the above limits.	A.1 Withdraw the sealed source from use.	Immediately
	AND	
	A.2.1 Decontaminate and repair the sealed source.	Immediately
	OR	
	A.2.2 Dispose of the sealed source in accordance with NRC Regulations	Immediately
	AND	
	A.3 Prepare and submit an annual report to the NRC for sealed sources or fission detectors that exceed the limits.	12 months

TESTING REQUIREMENTS

---NOTE---

Testing shall be performed by the Licensee or other persons specifically authorized by NRC or an Agreement State.

TEST		FREQUENCY
TR 16.12.2.1	------------------------------NOTE------------------------------ Only applicable for sources containing radioactive materials with half-lives greater than 30 days (excluding H3) and in any form other than gas. --- Each category of sealed source that is in use (excluding startup sources and fission detectors previously subjected to core flux) shall be tested for leakage and/or contamination with a detection sensitivity of at least 0.005 microCurie per test sample.	6 months
TR 16.12.2.2	------------------------------NOTE------------------------------ Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed in use. --- Each category of sealed source that is not in use (excluding startup sources and fission detectors previously subjected to core flux) shall be tested for leakage and/or contamination with a detection sensitivity of at least 0.005 microCurie per test sample.	Prior to use or transfer to another licensee unless tested in the previous 6 months
TR 16.12.2.3	Each sealed startup source or fission detector shall be tested for leakage and/or contamination with a detection sensitivity of at least 0.005 microCurie per test sample.	31 days prior to being subjected to core flux or installation in the core <u>AND</u> Following repair or maintenance to the source

BASES

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

REFERENCES

None

16.13 CONDUCT OF OPERATIONS

16.13.1 Fire Brigade

COMMITMENT A site Fire Brigade of at least five members shall be maintained onsite.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fire Brigade composition requirements not met.	A.1 Initiate action to fill required positions.	Immediately
	AND	
	A.2 Restore minimum fire brigade composition.	2 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.13.1.1 Verify fire brigade composition.	12 hours

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.

A fire brigade leader is required by the McGuire operating licenses and Fire Protection Program. The individual fulfilling this position shall:

- Have sufficient training or knowledge of plant safety related systems to understand the effects of a fire and fire suppression systems on safe shutdown capability,
- Be unavailable for other activities when directing the fire brigade,
- Be a licensed RO or SRO who is qualified to be a fire brigade leader.

The Fire Brigade requirement is met by using personnel from Operations and SPOC. Four (4) personnel from Operations are required (including the Fire Brigade Leader) and the other (1) person is from SPOC.

The Fire Brigade shall not include members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

Fire Brigade equipment and training conform to the recommendations of the National Fire Protection Association, Appendix A to Branch Technical Position 9.5-1 and supplemental NRC Staff guidelines.

This selected licensee commitment is part of the McGuire Fire Protection Program and therefore subject to the provisions of McGuire Facility Operating License Conditions C.4 (Unit 1) and C.7 (Unit 2).

REFERENCES

1. McGuire Nuclear Station UFSAR, Chapter 13.2.
2. McGuire Nuclear Station, SER Supplement 2, Chapter 9.5.1 and Appendix D.
3. McGuire Nuclear Station, SER Supplement 5, Chapter 9.5.1 and Appendix B.
4. McGuire Fire Protection Review, as revised.
5. McGuire Nuclear Station, SER Supplement 6, Chapter 9.5.1 and Appendix C.
6. McGuire Nuclear Station Facility Operating Licenses, Unit 1 License Condition C.(4) and Unit 2 License Condition C.(7)

16.13 CONDUCT OF OPERATIONS

16.13.2 Not Used

16.13 CONDUCT OF OPERATIONS

16.13.3 Not Used

16.13 CONDUCT OF OPERATIONS

16.13.4 Minimum Station Staffing Requirements

COMMITMENT Minimum station staffing shall be as indicated in Table 16.13.4-1.

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Minimum station staffing not met.	A.1 Initiate actions to fill required positions.	Immediately
	AND	
	A.2 Restore minimum station staffing.	2 hours

TESTING REQUIREMENTS

None

BASES

This SLC represents a consolidation of staffing requirements from a number of different regulatory requirements. The specific requirements that must be met at all times are described in the text below. The station must meet all of the regulatory requirements at all times. There is some interaction between the requirements associated with (1) the ability to mitigate design basis accidents and events, (2) fire events and (3) the emergency plan. The station must be able to demonstrate that the emergency plan is staffed as stated in the required emergency plan. During any design basis accident or event (including the design basis fire event) individuals from the emergency plan may fill mitigation roles for that purpose. However, the site must be able to demonstrate that no single individual would be called upon to perform conflicting tasks at any point in time during a design basis accident or event.

The requirements of this SLC consolidate McGuire station staffing requirements into one document. This SLC includes the station staff requirements of the McGuire Facility Operating Licenses, Technical Specification (TS) 5.2.2, 10 CFR 50.54(m), applicable Operations Management Procedures (OMPs), Nuclear System Directive (NSD) 112, "Fire Brigade Organization, Training and Responsibilities," the McGuire Fire Protection Program, the McGuire Emergency Plan, and SLC 16.13.1, "Fire Brigade." The total requirement for each position was obtained by summing the various individual requirements for that position. The bases for the numbers in the first column of Table 16.13.4-1 are as follows:

1 OSM (active SRO) – Required by 10 CFR 50.54(m)(2)(ii) and implemented via OMP.

1 STA (active or inactive SRO) – Required by TS 5.2.2g and implemented via OMP. Note that old TS (pre-Improved TS) Table 6.2-1, which implemented the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," did not require an STA on shift when both units were in MODE 5, 6, or defueled. Table 16.13.4-1 is more restrictive in that it requires an STA on shift at all times.

2 SROs (active SRO) – Required by 10 CFR 50.54(m)(2)(i). Per TS 5.2.2b and 10 CFR 50.54(m)(2)(iii), at least 1 SRO must be in the control room. Implemented by OMP.

4 ROs – Required by Tabletop Review of Abnormal Procedures, Memo to File dated 5/8/01. Implemented via OMP.

3 NLOs – Required by TS 5.2.2a and Section B, Figure B-1 of the Emergency Plan and implemented via OMP and NSD.

3 NLOs – Required by the Fire Protection Program and implemented via NSD and OMP.

1 Chemistry Technician (ERO) – Required by Section B, Figure B-1 of the Emergency Plan. Implemented via EP Group Manual Section 1.1. Any technician who is qualified may be credited towards fulfilling the ERO requirement.

3 Radiation Protection Technicians (2 technicians and 1 off-site dose assessor) (ERO) – Required by Section B, Figure B-1 of the Emergency Plan. Implemented via EP Group Manual Section 1.1. 1 technician is required by TS 5.2.2d and may be counted towards fulfilling the ERO requirement. Any technician who is qualified may be credited towards fulfilling the ERO requirement. In the event of a fire, the technician will respond to the fire for radiological monitoring purposes until directed otherwise.

1 Mechanical Maintenance Technician (ERO) – Required by Section B, Figure B-1 of the Emergency Plan. Implemented via EP Group Manual Section 1.1. Any technician who is fire brigade qualified may be credited towards fulfilling the ERO requirement and the fire brigade requirement. In the event of a fire, either the Mechanical Maintenance Technician or an Instrument and Electrical Technician will respond to the fire until directed otherwise.

2 Instrumentation and Electrical Technicians (ERO) – Required by Section B, Figure B-1 of the Emergency Plan. Implemented via EP Group Manual Section 1.1. Any technician who is fire brigade qualified may be credited towards fulfilling the ERO requirement and the fire brigade requirement. In the event of a fire, either the Mechanical Maintenance Technician or an Instrument and Electrical Technician will respond to the fire until directed otherwise.

2 MERT (ERO) – Required by Section B, Figure B-1 of the Emergency Plan. Implemented via EP Group Manual Section 1.1. Any technician who is qualified may be credited towards fulfilling the ERO requirement. In the event of a fire, the technician will respond to the fire for security purposes until directed otherwise.

BASES (con't)

Fire Brigade - The primary purpose of the Fire Protection Program is to minimize both the probability and consequence 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 5 members is required to provide immediate response to fires that may occur at the site. The fire brigade requirement is met by using personnel from Operations and SPOC. 4 personnel from Operations are required (including the fire brigade leader) and the other 1 person is from SPOC.

Fire Brigade Leader– Required by the McGuire Facility Operating Licenses and Fire Protection Program and implemented via NSD and OMP. The individual fulfilling this position shall be a SRO or RO who is qualified to be a fire brigade leader. This individual functions as the fire brigade leader and is not available for other activities when directing the fire brigade. The fire brigade leader shall have sufficient training in or knowledge of plant safety related systems to understand the effects of a fire and fire suppression systems on safe shutdown capability.

Minimum station staffing totals for the SRO, RO, and NLO positions in Table 16.13.4.1 are a function of the number of units in MODES 1-4. The totals for the remaining positions in Table 16.13.4.1 are not a function of the operational MODES of the units.

10 CFR 50.54(m)(2)(i) requires 2 SROs when both units are in MODES 1-4, 2 SROs when one unit is in MODES 1-4, and 1 SRO when no unit is in MODES 1-4.

10 CFR 50.54(m)(2)(i) requires 3 ROs when both units are in MODES 1-4, 3 ROs when one unit is in MODES 1-4, and 2 ROs when no unit is in MODES 1-4.

TS 5.2.2 a requires 3 NLOs when both units are in MODES 1-4, 3 NLOs when one unit is in MODES 1-4, and 2 NLOs when no unit is in MODES 1-4.

The 2-hour remedial action for restoring minimum station staffing levels is consistent with TS 5.2.2c and 5.2.2d, which allow 2 hours to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

TABLE 16.13.4-1

MINIMUM STATION STAFFING REQUIREMENTS

Title or Expertise	Both Units in MODES 1-4	One Unit in MODES 1-4	Both Units in MODES 5,6,or No MODE
Operations Shift Manager (OSM)	1	1	1
Shift Technical Advisor (STA)	1	1	1
Senior Reactor Operator (SRO) (Notes 1,2, 3)	2	2	1
Reactor Operator (RO) (Notes 1, 4)	4	4	3
Non-Licensed Operator (NLO)	6	6	5
Chemistry Technician	1	1	1
Radiation Protection	3	3	3
Mechanical Maintenance Technician	1	1	1
Instrumentation and Electrical Technician	2	2	2
Medical Emergency Response Team (MERT)	2	2	2
Security Personnel	Per Security Plan		
Fire Brigade	Per SLC 16.13.1		

TABLE 16.13.4-1

MINIMUM STATION STAFFING REQUIREMENTS (con't)

Note 1: Either a SRO (active or inactive) or a RO may be designated as the fire brigade leader. The totals for the appropriate position shall be increased by one, depending upon which position is being used to fulfill the role of fire brigade leader.

Note 2: In addition to these requirements, during CORE ALTERATIONS (including fuel loading or transfer), a SRO shall be present to directly supervise the activity. During this time, no other duties shall be assigned to this person.

Note 3: With any unit in MODES 1-4, a SRO shall be present in the control room at all times.

Note 4: For each fueled unit, a RO shall be present at the controls at all times.

REFERENCES

1. McGuire Facility Operating Licenses for Units 1 and 2, NPF-9 and NPF-17.

2. McGuire TS 5.2.2, "Unit Staff".

3. 10 CFR 50.54(m).

4. OMP 2-2, "Conduct of Operations."

5. NSD 112, "Fire Brigade Organization, Training and Responsibilities."

6. McGuire Emergency Plan.

7. SLC 16.13-1, "Fire Brigade."

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports For Nuclear Power Plants, LWR Edition," Section 9.5.1C3.

9. EP Group Manual Section 1.1, "Emergency Organization".

10. "Tabletop Review of AP/1&2/A/5500/24 (Loss of Plant Control Due to Fire Or Sabotage)" Memo to File dated May 8, 2001.

11. H. B. Barron letter to NRC Document Control Desk dated March 28, 2000, Emergency Plan Revision of Table B-1.

12. H. B. Barron letter to NRC Document Control Desk dated September 7, 2000, Emergency Plan Table B-1.

13. Frank Rinaldi to H. B. Barron dated October 3, 2000, McGuire Nuclear Station, Units 1 and 2 – Revision to Station Emergency Plan (TAC Nos. MA8908 and MA8909).

14. NRC Information Notice 95-48, "Results of Shift Staffing Study."

15. NRC Information Notice 91-77, "Shift Staffing at Nuclear Power Plants."

16. NRC SECY 93-184, "Shift Staffing at Nuclear Power Plants."

16.14 TESTING

16.14.1 Startup Reports

COMMITMENT

The following report shall be submitted in accordance with 10CFR50.4:

1. A summary report of plant STARTUP and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the 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 plant.

2. The Startup Report shall address each of the tests identified in the UFSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in License conditions based on other commitments shall be included in this report.

3. Startup Reports shall be submitted within: (1) 90 days following completion of the STARTUP test program, or (2) 90 days following resumption or commencement of commercial POWER OPERATION, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of STARTUP test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

APPLICABILITY

At all times.

REMEDIAL ACTION

None

TESTING REQUIREMENTS

Not Applicable.

BASES

This commitment is to satisfy the requirements of 10CFR50.4.

REFERENCES

1. Technical Specifications 6.9.1.1, 6.9.1.2, and 6.9.1.3 as amended through amendments 166/148.

2. UFSAR 14.0, Initial Tests and Operation, as revised through January 1, 1998.

16.14 TESTING

16.14.2 Reactor Trip Breaker and Solid State Protection System (SSPS) Logic Train Out of Service Commitments

COMMITMENTS: Risk-significant plant configurations shall not be entered when a reactor trip breaker or a SSPS logic train (Technical Specifications 3.3.1 and 3.3.2) is inoperable for maintenance:

1. To preserve ATWS mitigation capability, activities that degrade the ability of auxiliary feedwater system (AFWS), reactor coolant system (RCS) pressure relief systems (pressurizer PORVs and safety valves), ATWS mitigating system actuation circuitry (AMSAC), or turbine trip shall not be scheduled when a reactor trip breaker or a SSPS logic train is inoperable for maintenance.

2. To preserve LOCA mitigation capability, one complete Emergency Core Cooling System (ECCS) train that can be actuated automatically must be maintained when a SSPS logic train is inoperable for maintenance.

3. To preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable shall not be scheduled when a reactor trip breaker or a SSPS logic train is inoperable for maintenance.

4. Activities in electrical systems (e.g., AC and DC power) and cooling systems (e.g., essential service water and component cooling water) that support the systems or functions listed in Commitments 1,2 and 3 above shall not be scheduled when a reactor trip breaker or SSPS logic train is inoperable for maintenance. That is, one complete train of a function noted above must be available.

5. To preserve capabilities to prevent large early releases, activities that degrade the ability of the containment spray systems, air return fans, and ice condenser shall not be scheduled when a SSPS logic train is inoperable for maintenance.

APPLICABILITY: When a reactor trip breaker or a SSPS logic train is required to be operable per Technical Specification 3.3.1 or 3.3.2.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Risk-significant plant configurations identified in the COMMITMENTS are encountered during maintenance of a reactor trip breaker or a SSPS logic train.	A.1 Assess and manage the resulting risk in accordance with the Maintenance Rule program, 10 CFR 50.65(a)(4). AND	As soon as possible
	A.2 Exit risk-significant plant configurations.	As soon as possible

TESTING REQUIREMENTS

None

BASES

As specified in the NRC SERs dated December 30, 2008 and March 9, 2009 for License Amendments 248/228 and License Amendments 250/230, respectively, revising Technical Specifications 3.3.1 and 3.3.2, the commitments listed in this SLC are Tier 2 commitments.

Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed application, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that appropriate restrictions are in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed Completion Time is implemented.

Even though this SLC is designed to prevent entering risk-significant plant configurations, it is also applicable when risk-significant plant configurations are encountered during maintenance of a reactor trip breaker or a SSPS logic train.

Commitment 1 is associated with preserving ATWS mitigation capability. The phrase, "activities that degrade the ability", refers to those activities that make the listed mitigation systems, functions, or equipment, unable (i.e., unavailable) to perform their required functions.

Commitment 2 is associated with preserving LOCA mitigation capability. The phrase, "must be maintained", means the listed mitigation train function must be able (i.e., available) to perform the required function.

Commitment 3 is associated with preserving reactor trip and safeguards actuation capability on the train opposite the reactor trip breaker train or SSPS logic train that is inoperable for maintenance. The phrase, "activities that cause master relays or slave relays in the available train to be unavailable", refers to activities that make the master relay(s) or slave relay(s) in the train opposite the reactor trip breaker train or SSPS logic train that is inoperable for maintenance, unable (i.e., unavailable) to perform their required functions. The phrase, "activities that cause analog channels to be unavailable", refers to activities that cause any of the Reactor Protection System (RPS)/Engineered Safety Features Actuation System (ESFAS) analog channels that are inputs to SSPS to be unable (i.e., unavailable) to perform their required functions while either train of SSPS or any reactor trip breaker is inoperable for maintenance.

Commitment 4 is associated with activities in support systems that directly affect the availability of the systems and equipment listed in Commitments 1, 2 and 3. The systems and equipment listed in Commitments 1, 2 and 3 must be able (i.e., available) to perform their required functions. To ensure these systems and equipment are available, the required electrical systems (e.g., AC and DC power) and cooling systems that support these systems and equipment must also be available (i.e., able) to perform their required functions.

Commitment 5 is associated with preserving capabilities to prevent large early releases. The phrase, "activities that degrade the ability", refers to those activities that make the listed mitigation systems or equipment unable (i.e., unavailable) to perform their required functions.

The systems/functions and their associated TS/SLC involved in these commitments are:

- Reactor Trip Breakers – TS 3.3.1, Functions 17 and 18 (Commitments 1, 2, 3, 4, 5)
- SSPS Logic Train – TS 3.3.1 and TS 3.3.2 (Commitments 1, 2, 3, 4, 5)
- AFWS – TS 3.7.5 (Commitment 1)
- RCS Pressure Relief Systems (Pressurizer PORVs and Safety Valves) - TS 3.4.10, TS 3.4.11, and SLC 16.5.6 (Commitment 1)
- AMSAC – SLC 16.7.1 (Commitment 1)
- Turbine Trip – TS 3.3.2, Function 5 (Commitment 1)
- ECCS Train – TS 3.5.2 and TS 3.5.3 (Commitment 2)
- SSPS Master Relays, Slave Relays, and Analog Channels – TS 3.3.1 and TS 3.3.2 (Commitment 3)
- Electrical Systems (AC and DC) – TS 3.8.1, TS 3.8.3, TS 3.8.4, TS 3.8.6, TS 3.8.7, TS 3.8.9, and SLC 16.8.3 (Commitment 4)
- Cooling Systems (e.g., Essential Service Water (TS 3.7.7, TS 3.7.8) and Component Cooling Water (TS 3.7.6)) (Commitment 4)
- Containment Spray System – TS 3.6.6 (Commitment 5)

- Containment Air Return Fans – TS 3.6.11 (Commitment 5)

- Ice Condenser – TS 3.6.12 and TS 3.6.13 (Commitment 5)

REFERENCES

1. NRC SER dated 12/30/08, Issuance of Amendments 248/228.

2. NRC SER dated 3/9/09, Issuance of Amendments 250/230.

3. WCAP-15376-P-A, Rev. 1, Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times.

4. WCAP-14333-P-A, Rev. 1, Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times.

5. 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.