➔(EC-22790, Am. 119)

2.2.1 REACTOR TRIP SETPOINTS

BASES

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

🡐(EC-22790, Am. 119)

NOTE

The BASES contained in the succeeding pages summarize the reasons for the requirements of Sections 3.0 and 4.0.

→ (DRN 04-1191, Am. 91)

3/4.0 APPLICABILITY

BASES

LCO 3.0.1 establishes the Applicability statement within each individual LCO as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, communicate the situation with the General Manager - Plant Operations (GMPO) or his designee (e.g., Duty Plant Manager) as soon as practicable but no later than 7 hours; and initiate a condition report to document the condition and determine any limitations for continued operation of the plant. In some cases, when the ACTION is not completed within the allowed outage time, a shutdown may be required.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

LCO 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

← (DRN 04-1191, Am. 91)

LCO 3.0.3 establishes the requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this LCO is to communicate the situation to the General Manager – Plant Operations (GMPO) or his designee (e.g., Duty Plant Manager) as soon as practicable but no later than 7 hours and specify that a condition report must be generated when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. The GMPO notification assures plant management oversight and an appropriate level of conservative decision making will be applied in a timely manner. The condition report is required to be generated to document the condition and determine any limitations for continued operation of the plant.

The requirements of LCO 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

LCO 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance with these conditions would result in a possible shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a LCO by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements and any condition report limitations that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this LCO should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of LCO 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this LCO is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of Surveillance Requirements to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed Surveillance Requirements. This Specification does not provide time to perform any other preventive or corrective maintenance.
← (DRN 04-1191, Am. 91)

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the Surveillance Requirements.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of a Surveillance Requirement on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of a Surveillance Requirement on another channel in the same trip system.

SR 4.0.1 establishes the requirement that Surveillances must be performed during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual Surveillance Requirements. This specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified interval, in accordance with SR 4.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated Surveillance Requirements have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. The systems or components are known to be inoperable, although still meeting the Surveillance Requirements or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given Surveillance. In this case, the unplanned event may be credited as fulfilling the performance of the Surveillance. This allowance includes those Surveillances whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by LCO Action Statements do not have to be performed on inoperable equipment because the Action Statements define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 4.0.2, prior to returning equipment to OPERABLE status.
← (DRN 04-1191, Am. 91)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 4.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of SR 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

This extension allowed by SR 4.0.2 is also applicable to Surveillance Requirements required in TRM ACTIONS. However, the extension does not apply to the initial performance. The extension only applies to each performance after the initial performance. The initial performance required by the ACTION, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single completion time. One reason for not allowing the extension to this completion time is that such an action usually verifies that no loss of function has occurred or accomplishes the function of the inoperable equipment in an alternative manner.

SR 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within its specified interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 4.0.2, and not at the time that the specified interval was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with required actions or other remedial measures that might preclude completion of the Surveillance.
← (DRN 04-1191, Am. 91)

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with an interval based not on time intervals, but upon specified unit conditions, operational situations, or requirements of regulations is discovered to not have been performed when specified, SR 4.0.3 allows for the full delay period of up to the specified interval to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. SR 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by required actions.

Failure to comply with specified intervals for surveillance requirements is expected to be an infrequent occurrence. Use of the delay period established by SR 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified interval is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the allowed outage times of the required actions for the applicable LCO begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the allowed outage times of the required actions for the applicable LCO begin immediately upon the failure of the Surveillance.
← (DRN 04-1191, Am. 91)

Satisfactory completion of the Surveillance within the delay period allowed by this Specification, or within the allowed outage time of the actions, restores compliance with SR 4.0.1.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

SR 4.0.4 establishes the requirement that all applicable surveillance must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this SR, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Condition for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of SR 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.
← (DRN 04-1191, Am. 91)

→ (DRN 02-912)

3/4.1.2 BORATION SYSTEMS CHARGING PUMPS – APPENDIX R

← (DRN 02-912)

The Action of this TRM is required to be entered for charging pump A and charging pump B, separately. Also, the allowed outage time (AOT) of 7 days is tracked separately for each pump.

TRM 3/4.1.2 was developed to control the amount of time charging pumps A and B are allowed to remain inoperable. The Limiting Condition For Operation (LCO) requires charging pumps A and B to be Operable in Modes 1, 2, 3, and 4. The Applicability of Modes 1, 2, 3, and 4 is consistent with Technical Specifications (TS) 3.1.2.4 Applicability for the charging pumps. For the purposes of this TRM, a charging pump does not have to be aligned for service to be considered Operable; it only has to be readily available and capable of being aligned to perform its specified function.

The Action when the LCO cannot be met requires the inoperable charging pump(s) to be restored to Operable status within 7 days. If for some reason the inoperable charging pump(s) cannot be restored within 7 days, an hourly fire watch in the designated fire areas must be established within the next one hour. TRM Table 3.1-1, "Impacted Fire Areas with Charging Pump(s) Inoperable" lists the required fire areas that need fire watches when a required charging pump is inoperable.

The allowed outage time of 7 days is consistent with the "Remote Shutdown System" TS (TS 3.3.3.5). The requirement to establish an hourly fire watch compensates for not having one level of fire protection for greater than 7 days. The Waterford 3 fire hazards analysis relies on defense in depth for fire protection, but credits detection followed by timely extinguishing via the fire brigade.

No Surveillance Requirements are required per this TRM other than those required by Technical Specification 4.1.2.4.

Charging Pump AB is not required to be Operable for Appendix R purposes per this TRM. There are no fire areas that credit only charging pump AB as the Appendix R protected pump.

Technical Specification 3.1.2.4 Action requires a plant shutdown when the inoperable charging pump cannot be restored within 72 hours. This TRM requires a fire watch to be established if the pump is not restored within 7 days. If a TS required shutdown to Cold Shutdown is completed within 108 hours (72 hour AOT + 36 hours to CSD = 4.5 days), which is less than 7 days, fire watch patrols would not have to be deployed. However, if one pump is restored to Operable status (shutdown not required) and the other pump is inoperable for more than 7 days, fire watch patrols are required to be deployed.

→(DRN 03-2115, Am. 85; DRN 04-1244, Am. 99)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERING SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

Items 9 and 10 in Table 3.3-2 provide the surveillance test acceptance criteria for Core Protection Calculator (CPC) response times consisting of:

- CPC system hardware, plus
- reactor trip relay matrix, plus
- time to interrupt power to the CEA holding coils by means of the reactor trip switch gear (RTSG).

These are the components whose response times are to be measured in the surveillance procedures. With the sum of these measured response times less than the times shown in Table 3.3-2, the CPC system hardware, the reactor trip relay matrix and time to interrupt power to the CEA drive mechanism are demonstrated to be functioning to specification and will support the response times assumed in safety analyses.

Having established the functioning of the CPC hardware, the CPC trip delay times to be credited in the safety analyses for the various CPC system software functions are calculated based upon the maximum execution times, the execution schedules of the CPC software modules, the reactor trip relay matrix delay time and delay time to interrupt power to the CEA holding coils. The CPC trip delay times omit detector response times but include response times associated with processing the detector outputs. For example, they include the RTD bridge module delay time and the pump speed counter delay time.

The calculated safety analysis CPC trip delay times for the individual CPC System functions, corresponding to Table 3.3-2 instrumentation response time acceptance criteria, are listed below. Where different values are listed under Local Power Density trip and DNBR/Quality Margin trip, the higher of the two values is used in the Safety Analysis.

		Functional Unit	Delay Time (second(s))
9.		Local Power Density – High	
	a.	Neutron Flux Power From Excore Detectors	0.275
	b.	CEA Positions	1.304
	c.	CEA Positions: CEAC Penalty Factor #1	0.463
	d.	CEA Positions: CEAC Penalty Factor #2	0.463
10.		DNBR/Quality Margin – Low	
	a.	Neutron Flux Power From Excore Detectors	0.275
	b.	CEA Positions	1.343
	c.	Cold Leg Temperature	0.370[1]
	d.	Hot Leg Temperature	N/A[2]
	e.	Low DNBR Trip for Excess Load with Loss of A/C:	0.332[3]
	f.	Reactor Coolant Pressure from Pressurizer	0.270[4]
	g.	CEA Positions: CEAC Penalty Factor #1	0.502
	h.	CEA Positions: CEAC Penalty Factor #2	0.502

[1] Also applicable to CPC Auxiliary Trips on Differential Cold Leg Temperature and on low and high limits
[2] No Direct Trip on Hot Leg Temperature.
[3] A 0.332 second response time applies for this event, based on detection of a decrease of reactor coolant flow from conditions assumed to be at the power operating limit.
[4] Also applicable to CPC Auxiliary Trip on low and high limits

←(DRN 04-1244, Am. 99)

→(DRN 04-1244, Am. 99)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERING SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION (Continued)

The time delays acceptable to assume in accident and transient analysis for the CPC auxiliary trip delay times are:

Function	Delay Time (second(s))
Hot Leg Saturation trip:	2.952
Axial Shape Index range trip:	1.427
Integrated Radial Peaking Factor range trip:	1.427
Primary Coolant Pump Shaft Speed trip:	0.232
Differential Cold Leg Temperature (ASGT) trip:	0.370
Variable Over Power Trip (VOPT):	0.370[1]
Input range trip (Pressurizer pressure, Tcold)	0.370
Inadvertent RPC Event Trip	1.340[2]

[1] A 0.370 second response time is generally the maximum time assumption, where both temperature and neutron flux inputs to VOPT are considered.

[2] When the CPCS detects an RPC event, based on the RPC designated groups dropping into the core, it disables the processing of the target CEA inputs for a period of time controlled by an addressable constant, 18.5 seconds under EPU. After this time period, normal processing of the inputs is resumed. The Inadvertent RPC event will result in a trip if the core conditions so require after the CPC RPC timers complete their cycle, plus the delay stated here.

The time delay for several of the CPC trips can vary depending upon the dynamics of the various parameters which are input to the algorithms. Accident evolutions which affect inputs with sensitivity to dynamic parameter behavior are analyzed using CPCS simulation with the appropriate adjustments for the delays outside the CPCS, which are not included in the simulation. Such delays are due to the trip relay matrix, the RTSG and sensor lags.

←(DRN 04-1244, Am. 99)

←(DRN 03-2115, Am. 85)

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.2 INCORE DETECTORS

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

→(DRN 03-1608, Am. 83)

←(DRN 03-1608, Am. 83)

→(DRN 03-1608, Am. 83)

Limiting condition for operation (LCO) 3.3.3.2.d assures that misloaded fuel can be detected or adequate thermal margin is preserved for those misloads that cannot be detected. Either CEA symmetry testing, fuel symmetry verification testing, or an evaluation of the specific configuration of operable incore instruments must be performed prior to exceeding 50% power during initial startup testing for a cycle to detect potential fuel misloads that can have a significant impact on core peaking factors. If after successful completion of symmetry testing and during cycle operation, LCO 3.3.3.2.d requirements are not met, continued operation is allowed provided an evaluation of the specific configuration of operable incore instruments is performed and an appropriate penalty factor is installed in COLSS and CPCS. Seven days is acceptable to perform this evaluation since the core peaking factors and thermal margin requirements will change slowly with burnup.

←(DRN 03-1608, Am. 83)

3/4.3.3.3 SEISMIC INSTRUMENTATION

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. This 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,” April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

→(EC-1837, Am. 126)

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 initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, proposed Revision 1, September 1980, “Meteorological Programs in Support of Nuclear Power Plants”.

←(EC-1837, Am. 126)

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

→(DRN 04-1244, Am. 99)

←(DRN 04-1244, Am. 99)

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

→(EC-26965, Am. 120)

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and Surveillance 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. NRC Technical Specification Amendment 104 allowed the loose part detection instrumentation to be moved to the TRM.

←(EC-26965. Am. 120)

→(EC-5690, Am. 116)

The Neutron Noise Monitoring System and Safety Valve Position indication, although part of the Valve and Loose Parts Monitoring System, do not monitor for, nor detect impact of loose parts in the Reactor Coolant System, and as such are not subject to TRM 3.3.3.9, Loose Parts Detection Instrumentation.

←(EC-5690, Am. 116)

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

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 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 of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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 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 of 10 CFR Part 20. This instrumentation also include provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. 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.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification 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 overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment, or structures.

→(DRN 02-216)

NOTE: TRM Specification Bases 3/4.3.3.10 and 3/4.3.3.11 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

←(DRN 02-216)

➔(DRN 06-723, Am. 107)

3/4.3.4 TURBINE OVERSPEED PROTECTION (Cont'd)

Prevention of turbine overspeed and generation of potentially damaging turbine missiles is dependent upon proper valve function, overspeed detection and associated turbine runback or trip initiation and turbine disc integrity. Turbine missiles are created upon Low Pressure Turbine disc failure. While Low Pressure Turbine blade failure is an operational concern, Low Pressure Turbine blade failure is not an initiator of turbine missiles. Key factors to guard against generation of turbine missiles in addition to listed surveillances is to ensure Low Pressure Turbine discs remain clear of indications due to stress corrosion cracking, maintaining steam chemistry and preventing reductions in LP inlet temperature due to Moisture Seperator Reheater problems. Identification of indications in Low Pressure Turbine rotor discs due to stress corrosion cracking, operation with severe chemistry excursions for a period of time (days) or reductions in Low Pressure Turbine inlet temperatures due to Moisture Seperator Reheater problems for a period of time (days) are reasons to re-evaluate turbine rotor disc inspection intervals and turbine valve test intervals. Period of time (days) is based on discussion with Original Equipment Manufacturer Siemens (formerly Westinghouse) and Operating Experience.

🡐(DRN 06-723, Am. 107)

3/4.3.5 ULTRASONIC FLOWMETERS

➔(DRN 02-1889, Am. 74, DRN 03-247, Am. 75)

The ultrasonic flowmeters (UFMs) measure feedwater flow and bulk feedwater temperature. The UFM feedwater flow and feedwater bulk temperature inputs will be used by the Core Operating Limits Supervisory System (COLSS) to calculate station secondary calorimetric power. The UFM feedwater flow and feedwater bulk temperature inputs are also used as inputs into calibration constant algorithms that compensate or "calibrate" the alternate feedwater and main steam venturi-based flows and feedwater temperature instrumentation inputs used by COLSS on a loss of UFMs.

➔(DRN 04-1244, Am. 99)

The loss of a UFM will cause a control room alarm to annunciate and COLSS to automatically default to the compensated alternate venturi-based instrumentation inputs. COLSS normally defaults to Main Steam BSCAL (MSBSCAL) when reactor core power is greater than or equal to 95% of 3716 MWt RATED THERMAL POWER (RTP) or Feedwater BSCAL (FWBSCAL) when reactor core power is less than 95% of 3716 MWt RTP. MSBSCAL and FWBSCAL are calibrated by the UFM calibration factors. The requirement for the UFM to be operable above 50% power ensures that feedwater temperature is greater than the temperature (250 F) at which the UFM is reliable and the most accurate power measurement instrumentation is used over a large power range.

🡐(DRN 02-1889, Am. 74, DRN 03-247, Am. 75; DRN 04-1244, Am. 99)

➔(DRN 02-1889, Am. 74; DRN 03-247, Am. 75)

🡐(DRN 02-1889, Am. 74; DRN 03-247, Am. 75)

3/4.3.5 ULTRASONIC FLOWMETERS (cont'd)

➔(DRN 02-1889, Am. 74; DRN 03-247, Am. 75; DRN 04-1244, Am. 99)

Within 48 hours following the loss of the UFMs, operator action must be taken to reduce THERMAL POWER to less than or equal to 3697 MWt (99.5%). The decrease in power within the 48 hour completion time takes into account the reduction of confidence in the UFM based calibration factors resulting from COLSS alternate instrumentation loop drift caused by time and ambient temperature uncertainty effects. On restoration, THERMAL POWER should be maintained at the previous TRM action level until UFM calibration factors are developed.

←(DRN 02-1889, Am. DRN 03-247, Am. 75)

➔(DRN 03-247, Am. 75)

Within 31 days following the loss of the UFMs, operator action must be taken to reduce power to less than or equal to 3660 MWt (98.5%). The decrease in power within the 31 day completion time takes into account the loss of confidence in the UFM based calibration factors.

←(DRN 02-677, Am. 56; DRN 03-247, Am. 75; DRN 04-1244, Am. 99)

➔(DRN 02-1889, Am. 74; DRN 03-247, Am. 75)

The 48 hour and 31 day LCO ACTION STATEMENTS are required to maintain consistency with the COLSS Secondary Calorimetric Measurement Uncertainty analyses. The entry into LCO ACTION STATEMENTS begins with the loss of the UFM, when greater than 50% power. The appropriate ACTION STATEMENT time limit entry will be based on the last time MSBSCAL and FWBSCAL were updated by the UFM calibration factors prior to UFM failure. These ACTIONS ensure CPC margins to trip remain conservative and preserve the Appendix K ECCS limits.

←(DRN 02-1889, Am. 74; DRN 03-247, Am. 75)

➔(DRN 03-247, Am. 75; DRN 04-1244, Am. 99)

If COLSS is out of service, then Core Protection Calculator System (CPCS) will continue to maintain plant operations within the core power operating limits. Operating limits will be maintained through compliance with Technical Specifications (TS) sections 3.2.1, 3.2.3, 3.2.4, 3.2.7, and 4.3.1.1, Table 4.3-1 (2, 9, 10, 14) applicable ACTION STATEMENTS and Surveillance Requirements (SR). If the UFM(s) is OPERABLE during the period COLSS is out of service, then plant operation may continue at 3716 MWt RTP using the power indications from the CPCS and UFM based manual secondary calorimetric measurement. If the UFM(s) becomes INOPERABLE during the period COLSS is out of service, then plant operation may continue at 3716 MWt RTP using the power indications from the CPCS. However, in order to remain in compliance with the bases for operation at a RTP of 3716 MWt, the UFM(s) must be returned to service prior to the next required daily CPCS calibration or THERMAL POWER must be reduced to less than or equal to 3660 MWt (98.5%). This power reduction is performed prior to the next CPC calibration in order to remain within the alternate venturi-based instrumentation power measurement uncertainty analysis and maintain consistency with the COLSS Secondary Calorimetric Measurement Uncertainty analyses.

←(DRN 03-247, Am. 75; DRN 04-1244, Am. 99)

3/4.3.5 ULTRASONIC FLOWMETERS (cont'd)

The TRM is annotated with a 3.0.4 exemption, allowing entry into the applicable Mode to be made with UFMs INOPERABLE, as required by the Actions.

The SR 4.3.5.a. to perform a CHANNEL FUNCTIONAL TEST at least once per 18 months is based on the vendor recommendations. The UFM equipment contains on-line self-diagnostic capabilities to continuously verify operation within its design bounds. The 18 month frequency is based on the refueling cycle. This frequency is acceptable from a reliability standpoint.

The SR 4.3.5.b 31 day CHANNEL CHECK verifies the COLSS alternate calorimetric heat balances MSBSCAL and FWBSCAL are within a value bounded by engineering analyses. This comparison of alternate heat balances to USBSCAL assures the calibration factors derived by USBSCAL are valid and the basis for the original power measurement uncertainty assumptions for operation are maintained at the various thermal power levels when UFM is inoperable.
←(DRN 02-677, Am. 56)

➔(DRN 04-1244, Am. 99)

3/4.4.3 PRESSURIZER

The heater capacity cited in requirement 3.4.3.1 is sufficient, in conjunction with the heater capacity required by Technical Specification 3.4.3.1b, to bound the capacity credited in the analysis of the CEA Withdrawal within Deadband event. The additional heater capacity cited in requirement 3.4.3.1 can be heaters powered from any combination of Class 1E or non-class 1E buses.

←(DRN 04-1244, Am. 99)

➔(DRN 07-201, Am. 112)

3/4.4.5 LEAKAGE DETECTION INSTRUMENTATION

Background

This specification is applicable to the containment gaseous radioactivity monitor and the containment fan cooler condensate flow switches and is based upon the regulatory requirements in Reg. Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems. The position of Reg. Guide 1.45 is that at least three different leakage detection methods should be employed. The containment gaseous radioactivity monitor and the containment fan cooler condensate flow switches constituted the third leakage detection method and were contained in the original Technical Specification 3.4.5.1 as the third Reactor Coolant System (RCS) Leakage Detection system.

With the improved integrity of the reactor fuel and the resultant reduced RCS radioactivity levels, the containment gaseous radioactivity monitor became less effective for RCS leakage detection. The containment gaseous radioactivity monitor may not detect a one gallon per minute Reactor Coolant System leak within one hour. In Amendment 197, the NRC granted removal of the containment gaseous radioactivity monitor from Technical Specification 3.4.5.1 due to the equipment inability to detect a one gallon per minute Reactor Coolant System leak within one hour.

Subsequently, it was recognized that the containment fan cooler condensate flow switches were not capable of detecting a one gallon per minute Reactor Coolant System leak within one hour. The containment fan cooler condensate flow switches monitor condensate flow from individual containment fan coolers and do not sum the condensate flow from all containment fan coolers. The containment fan cooler condensate flow switches assist in detecting the source of leakage into the containment sump. In Amendment 212, the NRC granted removal of the containment fan cooler condensate flow switches from Technical Specification 3.4.5.1 due to the equipment inability to detect a one gallon per minute Reactor Coolant System leak within one hour. In the Amendment Request, Waterford 3 committed to maintaining the containment gaseous radioactivity monitor and the containment fan cooler condensate flow switches functional and available.

←(DRN 07-201, Am. 112)

In the Safety Evaluation of Amendment 212, the NRC acknowledged that the Licensee utilized four separate methods of Reactor Coolant System leakage detection instrumentation, but that only two of the instruments meet the sensitivity requirements of Regulatory Guide 1.45 (i.e. of being able to detect a one gallon per minute Reactor Coolant System leak within one hour). Further, the NRC acknowledged that deletion of the containment fan cooler condensate flow switches from Technical Specification 3.4.5.1 is an exception to the recommendations of Regulatory Guide 1.45. In the FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, the NRC credited Waterford 3 with maintaining the containment gaseous radioactivity monitor and the containment fan cooler condensate flow switches FUNCTIONAL and AVAILABLE in the justification of two out of the three determination questions.

Limiting Condition for Operation

The ACTION requirement to initiate a Condition Report within the next 6 hours after the third Reactor Coolant System (RCS) Leakage Detection system has not been OPERABLE for greater than 72 hours provides Waterford 3 sufficient time to perform an initial review to determine whether the third RCS Leakage Detection system is available to perform its function (i.e. of providing indication of Reactor Coolant System leakage to enable Operations to promptly identify Reactor Coolant Pressure Boundary Leakage and to the extent practical, locate the source of the leakage).

The containment fan cooler condensate flow switches on the operable Containment Fan Coolers is based upon the Technical Specification 3.6.2.2 requirement that two trains of Containment Cooling shall be operable with one fan cooler unit to each train. The operable Containment Fan Coolers must be in operation for the condensate flow switches to effectively perform their function of Reactor Coolant System leakage detection. For the Containment Fan Coolers that have two condensate flow switches per unit (i.e. Containment Fan Coolers A and C) only one condensate flow switch is required to be operable per unit to perform the Reactor Coolant System leakage detection function.

Surveillance Requirements

The Surveillance Requirement 4.4.5.1.a for performance of a 12 hour CHANNEL CHECK of the containment gaseous radioactivity monitor ensures that the monitor can perform its function in the desired manner. The check gives reasonable confidence the channel is operating properly. The frequency of 12 hours is based upon instrument reliability and is reasonable for detecting off normal conditions.

The Surveillance Requirement 4.4.5.1.a for performance of a quarterly CHANNEL FUNCTIONAL TEST of the containment gaseous radioactivity monitor ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The frequency of quarterly performance considers instrument reliability, and operating experience has shown it proper for detecting degradation.
← (DRN 007-201, Am. 112)

(DRN 04-1244, Am. 99)

The Surveillance Requirement 4.4.5.1.a for performance of an 18 month CHANNEL CALIBRATION TEST of the containment gaseous radioactivity monitor verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this frequency is acceptable.

The Surveillance Requirement 4.4.5.1.b for performance of an 18 month CHANNEL FUNCTIONAL TEST of the containment fan cooler condensate flow switches verifies the alarm function by providing water flow through each containment fan cooler drain line. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this frequency is acceptable.

(DRN 04-1244, Am. 99; DRN 07-201, Am. 112)

(LBDCR 13-003, Am. 124)

3/4.4.6 CHEMISTRY

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 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 Steady State Limits.

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

(LBDCR 13-003, Am. 124)

3/4.4.8 PRESSURE TEMPERATURE LIMITS

The limitations imposed on the pressurizer heatup and cooldown rates are provided to assure that the pressurizer is operated within the design criteria assumed for fatigue analysis performed in accordance with ASME Code requirements.

(DRN 04-1238, Am. 92)

3/4.4.9 STRUCTURAL INTEGRITY

(DRN 06-588, Am. 108)

This specification is applicable to the Reactor Coolant Pressure Boundary (RCPB). The RCPB, as defined by 10 CFR 50.2 and FSAR Section 5.2, Integrity of Reactor Coolant Pressure Boundary, includes all pressure containing components such as pressure vessels, piping, pumps, and valves which are:

(DRN 03-1808, Am. 80; DRN 04-1238, Am. 92; DRN 06-588, Am. 108)

3/4.4.9 STRUCTURAL INTEGRITY (cont'd)

1. Part of the Reactor Coolant System (RCS); or
2. Connected to the RCS, up to and including any and all of the following:
   a. The outermost containment isolation valve in system piping that penetrates the primary containment.
   b. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the primary containment.
   c. The RCS primary safety valves.

←(DRN 04-1238, Am. 92)

Refer to the Waterford 3 Inservice Inspection (ISI) Program and FSAR sections 5.2 and 6.6 for a detailed description of the applicable systems/subsystems that make up the RCPB. Structural integrity of systems outside the boundaries of the RCPB will be maintained through compliance with 10 CFR 50.55a, as implemented through the Waterford 3 ISI program.

←(DRN 06-588, Am. 108)

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 10CFR50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10CFR50.55a(g)(6)(i).

The structural integrity of components means the vessels, piping systems, pumps, valves and component supports are of a state of quality such that they are complete, whole, entire and unbroken. Structural integrity requirements are not applicable to non-structural components such as packing, gaskets, o-rings, etc., and other sealing devices. TRM 3.4.9 is applicable to ASME Code Class 1, 2, and 3 components during all MODES in which the

system, of which the components are a part, is required to be OPERABLE. TRM 3.4.9 does not apply to systems unless the systems are required to be OPERABLE by some other specification. If the system is not required to be OPERABLE, then the components are not required to be OPERABLE and the structural integrity of the component is not required to be maintained. This TRM is not intended to track the structural integrity of the systems that may be disassembled or repaired during an outage.

TRM 3.4.9 is applicable to systems or portions of systems that are performing a support function for systems that are required to be OPERABLE even if the supporting system is not required to be OPERABLE by a specification. When a support system, or a system performing a support function is not required to be OPERABLE but is required to perform its support system or function role for a system that is required to be OPERABLE then the components performing the support function must have structural integrity and TRM 3.4.9 would apply to the portion of the system and the related components performing the support function. System components that are not required to perform the support function would not be required to have structural integrity and TRM 3.4.9 would not apply.

←(DRN 03-1808, Am. 80)

3/4.4.9 STRUCTURAL INTEGRITY (cont'd)

→DRN 03-1808, Am. 80)

The ACTION requirements provide restrictions prior to restoring structural integrity on ASME Code Class 1 and 2 components while allowing sufficient Reactor Coolant System heatup to allow pressurization to perform hydrostatic testing of the affected component while complying with the RCS pressure/temperature limits of Specification 3.4.8.1. For the structural integrity of the reactor coolant pressure boundary to be assured, the reactor coolant pressure boundary must be maintained within the limitations of Technical Specifications 3.4.8.1 and 3.4.8.2. Maintaining the reactor coolant pressure boundary within these limits ensures that the component material is adequately warmed such that the material is above the Nil Ductility Transition Temperature prior to introducing pressure sufficient to induce brittle fracture. The Nil Ductility Temperature of the reactor vessel is 22°F plus adjustments for embrittlement when the reactor coolant system pressure is greater than 525 psi. At pressure below 525 psi, lower temperatures are acceptable. These limits are controlled by the Pressure/Temperature limits in Technical Specification 3.4.8.1.

The lowest service temperature per ASME is NDT plus 60°F for the reactor vessel and NDT plus 100°F for the piping, pumps and valves. The most limiting NDT of the reactor coolant system is 90°F based on the reactor coolant pump driver mount flanges. That makes the lowest service temperature 215.6°F (90°F plus 100°F plus instrument error). The minimum temperatures apply

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, 1980 Edition and Addenda through Winter 1981.

←(DRN 03-1808, Am. 80)

3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance requirements for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50. A one time extension of the test interval is allowed for the third Type A test of the first 10-year service period, as required by Surveillance Requirement 4.6.1.2.a and by Section III.D.(a) of Appendix J to 10 CFR Part 50, provided the performance of the Type A test occurs prior to unit restart following Refuel 7.

→(DRN 04-1244, Am. 99)

←(DRN 04-1244, Am. 99)

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

→ (DRN 03-667, Am. 77)

Table 3.6-2, notation 8, allows the opening of closed containment isolation valves on an intermittent basis under administrative controls. The valves within the scope of this footnote include locked or sealed closed containment isolation valves and deactivated automatic containment isolation valves secured in the isolation position. Acceptable administrative controls must include the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

← (DRN 03-667, Am. 77)

→ (DRN 04-972, Am. 90)

3/4.6.4 COMBUSTIBLE GAS MONITORING

The OPERABILITY of the equipment and systems required for the detection of hydrogen gas ensures that this equipment will be available to assess the degree of core damage during a beyond design basis accident and confirm that random or deliberate ignition has taken place. If an explosive mixture that could threaten containment integrity exists during a beyond design-basis accident, then severe accident management strategies, such as purging and/or venting, would need to be considered. The hydrogen monitors are needed to implement these severe management strategies.

The hydrogen monitors are not required to mitigate design-basis accidents and, therefore, do not meet the definition of a safety-related component as defined in 10 CFR 50.2. As part of the rulemaking to revise 10 CFR 50.44, the Commission found that the hydrogen monitors no longer met the definition of Category 1 in Regulatory Guide 1.97. The Commission concluded that Category 3, as defined in Regulatory Guide 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. Hydrogen monitoring is not the primary means of indicating a significant abnormal degradation of the reactor coolant pressure boundary.

← (DRN 04-972, Am. 90)

3/4.7.1.6.1 MAIN FEEDWATER REGULATING VALVES AND STARTUP FEEDWATER REGULATING VALVES

The Main Feedwater Isolation Valves (MFIVs) isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break. Closure of the MFIVs terminates flow to both steam generators, terminating the event for feedwater and steam line breaks occurring downstream of the MFIVs. The consequences of events occurring in the MFW lines upstream of the MFIVs will also be mitigated by their closure.

Main Feedwater Regulating Valves (MFRV) and Startup Feedwater Regulating Valves (SFRV) are located upstream of the MFIV and are used as a backup to their related MFIV. Both valves must close to isolate the line. These valves close on a MSIS and are furnished with emergency closure circuits so that the closure can be actuated through override of their normal control signals.

In MODES 1, 2, 3, and 4, the SFRVs and MFRVs must be capable of actuating to the closed position on a MSIS, except when they are closed and deactivated or isolated by either a closed manual valve or closed and deactivated automatic valve. When a SFRV or MFRV is closed and deactivated or isolated by a closed manual valve or closed and deactivated automatic valve, it is already performing its safety function and continued operation in the applicable MODES is allowed.

With one SFRV or MFRV unable to actuate to the closed position on a MSIS, action must be taken to close or isolate the inoperable valve within 72 hours. The 72 hour Completion Time is consistent with that for the MFIVs. The 72 hour Completion Time is reasonable to return the SFRVs and MFRVs to functional status or isolate the affected flow path.

SFRVs and MFRVs that are closed or isolated because they are unable to actuate to the closed position on a MSIS must be verified on a periodic basis that they are closed or isolated. The 7 day time is reasonable in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated.

If a non-functional SFRV and MFRV cannot be repaired, closed, or isolated in the required time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 in the following 30 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

The TRM requirement is annotated with a 3.0.4 exemption, allowing entry into the applicable MODEs to be made with a SFRV or MFRV closed or isolated as required by the ACTIONs. The ACTIONS allow separate condition entry for each valve by using "With one or more SFRV or MFRV...". This prevents immediate entry into TS 3.0.3 if multiple SFRVs and/or MFRVs are unable to actuate to the closed position on a MSIS.

3/4.7.1.6.1 MAIN FEEDWATER REGULATING VALVES AND STARTUP FEEDWATER REGULATING VALVES (Continued)

→(DRN 03-1808, Am. 80)

The Surveillance Requirement to verify isolation in less than or equal to 4.5 seconds supports the 5.0 second time assumed in the accident and containment analyses minus 0.5 seconds to account for measurement uncertainty. The SFRVs and MFRVs should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. The Surveillance to verify each SFRV and MFRV can close on an actual or simulated actuation signal is normally performed when the plant is returning to operation following a refueling outage. The 18 month frequency is based on the refueling cycle. Verification of closure time is performed per the Inservice Testing Program. This frequency is acceptable from a reliability standpoint and is in accordance with the Inservice Testing Program.

←(DRN 03-1808, Am. 80)

→(DRN 02-1683)

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3/4.7.3 COMPONENT COOLING WATER SYSTEM - APPENDIX R

The Action of this TRM is required to be entered for component cooling water (CCW) pump A and CCW pump B, separately. Also, the allowed outage time (AOT) of 7 days is tracked separately for each pump.

TRM 3/4.7.3 was developed to control the amount of time CCW pumps A and B are allowed to remain inoperable. The Limiting Condition For Operation (LCO) requires CCW pumps A and B to be Operable in Modes 1, 2, 3, and 4. The Applicability of Modes 1, 2, 3, and 4 is consistent with Technical Specifications (TS) 3.7.3 Applicability for the CCW pumps. For the purposes of this TRM, a CCW pump does not have to be aligned for service to be considered Operable; it only has to be readily available and capable of being aligned to perform its specified function.

The Action when the LCO cannot be met requires the inoperable CCW pump(s) to be restored to Operable status within 7 days. If for some reason the inoperable CCW pump(s) cannot be restored within 7 days, an hourly fire watch in the designated fire areas must be established within the next one hour. TRM Table 3.7-1, "Impacted Fire Areas with CCW Pumps Inoperable" lists the required fire areas that need fire watches when a required CCW pump is inoperable.

The allowed outage time of 7 days is consistent with the "Remote Shutdown System" TS (TS 3.3.3.5). The requirement to establish an hourly fire watch compensates for not having one level of fire protection for greater than 7days. The Waterford 3 fire hazards analysis relies on defense in depth for fire protection, but credits detection followed by timely extinguishing via the fire brigade.

No Surveillance Requirements are required per this TRM other than those required by Technical Specification 4.7.3.

CCW Pump AB is not required to be Operable for Appendix R purposes per this TRM. There are no fire areas that credit only CCW pump AB as the Appendix R protected pump.

Technical Specification 3.7.3 Action requires a plant shutdown when the inoperable CCW pump cannot be restored within 72 hours. This TRM requires a fire watch to be established if the pump is not restored within 7 days. If a TS required shutdown to Cold Shutdown is completed within 108 hours (72 hour AOT + 36 hours to CSD = 4.5 days), which is less than 7 days, fire watch patrols would not have to be deployed. However, if one pump is restored to Operable status (shutdown not required) and the other pump is inoperable for more than 7 days, fire watch patrols are required to be deployed.
🡐(DRN 02-1794)

➔(LBDCR 13-003, Am. 124; LBDCR 13-001, Am. 125)
3/4.7.5 FLOOD PROTECTION

The Limiting Condition for Operation provides a limitation on flood protection that ensures facility protective actions will be taken in the event of flood conditions. The limit of elevation +27.0 ft Mean Sea Level is based on the maximum elevation at which the levee provides protection. The nuclear plant island structure provides protection to safety-related equipment up to elevation +30 ft Mean Sea Level. The Action requirement to initiate and complete procedures ensuring that all doors and penetrations to the nuclear island are secure prior to attaining +27.0 ft Mean Sea Level establishes the nuclear plant island structure as flood protection for safety-related equipment. EC41353 determined a conservative response time to close the doors and EC39741 determined a maximum expected rate of rise in the Mississippi River using
🡐(LBDCR 13-003, Am. 124; LBDCR 13-001, Am. 125)

➔(LBDCR 13-001, Am. 125)

3/4.7.5 FLOOD PROTECTION (continued)

historical data. Based on the information documented in EC41353 and EC39741, the action to close the doors can be completed with 12 hours. Using a 12 hour projection of River Level will ensure that the flood protective actions to close the doors and penetrations are achieved. The surveillance requirements to measure the water level at the levee fronting the Waterford Unit 3 site provides information to be used to establish the required configuration for the affected doors and penetrations as described in the Action statement.

EC39741 determined that the Mississippi River level would increase form 24 ft. to 27 ft. in 40 hours based on historical data. The surveillance requirement to ensure water level at the levee fronting the Waterford Unit 3 site at least once per 24 hours when the water level is equal to or above elevation +24.0 ft. Mean Sea Level USGS falls within the 40 hour historical rise and provides a data point for determining if there is a need for increased monitoring.

🡐(LBDCR 13-001, Am. 125)

3/4.7.8 Shock Suppressors (Snubbers)

BACKGROUND

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

Snubbers required to be operable ensure that the structural integrity of the reactor coolant system and 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 non-safety 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.

ACTIONS

Because this TRM envelopes snubbers located in various locations, the ACTIONS are modified by Note 1 permitting separate condition entry for each inoperable snubber.

Note 2 captures the requirements of Technical Specification Task Force (TSTF) 372, such that the Emergency Feedwater (EFW) train not associated with the system train on which the inoperable snubber(s) resides must be available in any MODE where EFW is required to be OPERABLE in accordance with TS 3.7.1.2. During MODES where EFW is not required to OPERABLE per TSs, an alternate means of core cooling train not associated with the system train on which the inoperable snubber(s) resides must be available. These “opposite train” requirements are applicable only when an inoperable snubber(s) affects a single train of a given system. When the inoperable snubber(s) affects both trains of a given system, either EFW train or alternate means of core cooling train must be available, again, dependent on the MODES in which TSs require OPERABILITY of these systems. Refer to TS LCO 3.0.8 Bases for further information with respect to the intent of Note 2.

🡐(EC-15515, Am. 118)

ACTIONS (continued)

Note 3 states that ACTIONS b and c are not required to be entered, or may be exited if already entered, if an engineering evaluation has been performed and determined that the snubber(s) is not required for continued OPERABILITY of the supported system. Since Table 3.7.8-2 snubbers have both a dynamic and seismic requirement, the engineering evaluation would need to determine that neither of these features is required for continued OPERABILITY of the supported system in order to avoid entry into ACTION c (or exit ACTION c, as applicable). Based on such an evaluation, the snubber would no longer be "required" for supported system OPERABILITY. Inoperable snubbers that have been determined to not render the supported system inoperable may be removed from the associated TRM table, if desired, in accordance with 10 CFR 50.59. However, because individual and various combinations of snubbers can have differing impacts on supported system OPERABILITY, and because safety-related snubbers are generally required to be tested in accordance with TRM 6.5.1, it may be desirable to maintain these snubbers within the associated TRM 3.7.8 table.

If a snubber(s) is determined to not be required for supported system OPERABILITY and is not removed from the associated TRM table in accordance with 10 CFR 50.59, then a separate evaluation will be required for future inoperabilities of the snubber(s). Future evaluations may be based on, and refer to, the previous evaluation. These future evaluations are required since system configuration or operation can change over time, along with regulatory requirements for support systems or components such as snubbers.

When Note 3 is applied, the snubber(s) may be restored in a timeframe commensurate with the importance to safety. However, ACTION a may result in TS LCO entry regardless of the importance of the snubber(s).

ACTION a

When a snubber(s) is found to be inoperable, in addition to the determination of the snubber mode of failure, an assessment of impact on any safety related component or system that has been adversely affected by inoperability of the snubber(s) must be performed. The engineering evaluation is performed to determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system. This ACTION might be entered when a snubber is noted to be inoperable while installed on the system or found to be inoperable during bench testing following removal from the system. This possible impact to system OPERABILITY is applicable to both snubbers having dynamic and seismic features, and to seismic-only snubbers. If the engineering evaluation determines that system damage has occurred such that the system should be declared inoperable, then the associated TS LCO should be entered immediately.

ACTION b

Snubbers having only a seismic-related support system function are listed in TRM Table 3.7.8-1. When any snubber(s) in this table is found inoperable, TS LCO 3.0.8 is immediately applicable (provided the EFW or alternate means of core cooling requirements of Note 2 above are met). Depending on the snubber(s) location, TS LCO 3.0.8 may permit 72 hours or 12 hours for returning the snubber(s) to OPERABLE status.

In accordance with TSTF 372, TS LCO 3.0.8 cannot be applied when the requirement for EFW or alternate core cooling means is not met. In such cases, the requirements of ACTION b cannot be met and ACTION c must be entered.
🡐(EC-15515, Am. 118)

ACTIONS (continued)

In addition to the above, configuration risk management must be maintained when a required snubber(s) is inoperable. The risk is evaluated in accordance with the Risk Management Guidelines. These guidelines provide direction should station risk become sufficiently elevated during periods of snubber(s) inoperability. Direction may include establishment of compensatory measures or a dedicated effort to restore equipment to an OPERABLE status in order to reduce the overall station risk.

ACTION b.3 is discussed above (see ACTIONS Note 3 Bases). Note that ACTION b.3 is not required to be performed unless desired to determine if TS LCO 3.0.8 can be exited without restoration of the snubber(s).

ACTION c

Snubbers having both a dynamic and seismic related support system function are listed in TRM Table 3.7.8-2. When any snubber in this table is found inoperable, the TS LCO associated with the affected supported system must be entered. This is because TS LCO 3.0.8 cannot be applied to snubbers having a dynamic feature required for supported system OPERABILITY.

If an evaluation is performed and it is found that the supported system remains OPERABLE from both a dynamic and seismic perspective given the inoperable snubber(s), then the associated TS LCO and TRM 3.7.8 ACTION may be exited. However, there may be cases where the supported system is determined to be OPERABLE with respect to the inoperable dynamic feature of the snubber(s), but remains inoperable with respect to the inoperable seismic feature of the snubber(s). In such cases, the associated TS LCO for the supported system may be exited and TS LCO 3.0.8 applied. In accordance with Note 5, the TS LCO 3.0.8 entry must be applied from the time the snubber(s) was initially declared inoperable. If the time permitted by TS LCO 3.0.8 for restoration has already expired, then the unit should remain in the associated TS LCO for the supported system.

If the snubber(s) is not restored to OPERABLE status within the time provided by TS LCO 3.0.8, then the supported system must be declared inoperable and the associated TS LCO entered.

The required risk assessment is equivalent to that described for ACTION b.2 above.

In accordance with Note 4, if ACTION c was entered due to the inability to meet the requirements of ACTION b OR if EFW (or alternate means of core cooling) is unavailable in accordance with Note 2, then ACTIONS c.3, c.4, c.5, and c.6 are not applicable. In such cases, the supported system TS will govern ACTIONS necessary for the condition. Refer to TS LCO 3.0.8 Bases for additional information.

TEST REQUIREMENTS

TR 4.7.8.1 and TR 4.7.8.2

Visual inspections and functional tests of snubbers will be performed in accordance with the ASME OM Code. Subsection ISTD includes rules for visual inspection and functional testing of dynamic restraints (snubbers). These rules are based on maintaining a constant level of protection to plant systems from dynamic shocks such as may occur during earthquakes or transient dynamic system events. Code Case OMN-13 establishes specific requirements that must be met in order to allow extension of the visual examination interval beyond the maximum interval allowed in Table ISTD-4252-1 for mechanical and hydraulic snubbers.
←(EC-15515, Am. 118)

(LBDCR13-003, Am. 124)

3.7 PLANT SYSTEMS

TEST REQUIREMENTS (continued)

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alphaemitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage frombyproduct, 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 frequency 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 shield mechanism.

(LBDCR13-003, Am. 124)

3/4.7.10 FIRE SUPPRESSION SYSTEMS

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, fire hose stations, and yard fire hydrants. 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 Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

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.

3/4.7.11 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM CHILLERS - APPENDIX R

The Action of this TRM is required to be entered for each train of chilled water (CHW) chiller and associated pump separately. Also, the allowed outage time (AOT) of 7 days is tracked separately for each chiller and/or associated pump.

TRM 3/4.7.12 was developed to control the amount of time the CHW chillers and associated pumps are allowed to remain inoperable. The Limiting Condition For Operation (LCO) requires the CHW chillers and associated pumps to be Operable in Modes 1, 2, 3, and 4. The Applicability of Modes 1, 2, 3, and 4 is consistent with Technical Specifications (TS) 3.7.12 Applicability for the CHW chillers and associated pumps. For the purposes of this TRM, a CHW chiller and/or associated pump does not have to be aligned for service to be considered Operable; they only have to be readily available and capable of being aligned to perform their specified function.

The Action when the LCO cannot be met requires the inoperable CHW chiller(s) and/or associated pump(s) to be restored to Operable status within 7 days. If for some reason the inoperable CHW chiller(s) and/or pump(s) cannot be restored within 7 days, an hourly fire watch in the designated fire areas must be established within the next one hour. TRM Table 3.7-12, “Impacted Fire Areas with CHW Pumps or
Chillers Inoperable” lists the required fire areas that need fire watches when a CHW pump or chiller is inoperable.

The allowed outage time of 7 days is consistent with the “Remote Shutdown System” TS (TS 3.3.3.5). The requirement to establish an hourly fire watch compensates for not having one level of fire protection for greater than 7 days. The Waterford 3 fire hazards analysis relies on defense in depth for fire protection, but credits detection followed by timely extinguishing via the fire brigade.

No Surveillance Requirements are required per this TRM other than those required by Technical Specification 4.7.12.

Technical Specification 3.7.12 Action requires a plant shutdown when the inoperable CHW chiller and/or associated pump can not be restored within 72 hours. This TRM requires a fire watch to be established if the pump(s) and/or chiller(s) can not be restored within 7 days. If a TS required shutdown to Cold Shutdown is completed within 108 hours (72 hour AOT + 36 hours to CSD = 4.5 days), which is less than 7 days, fire watch patrols would not have to be deployed. However, if one pump and/or chiller is restored to Operable status (shutdown not required) and the other pump and/or chiller is inoperable for more than 7 days, fire watch patrols are required to be deployed.

3/4.7.13 SWITCHGEAR AREA VENTILATION SYSTEM

→ (DRN 02-1609)

The OPERABILITY of the switchgear area ventilation system ensures that sufficient cooling is supplied to spaces containing electrical equipment located in the reactor auxiliary building cable vault and switchgear areas required for safety-related operations and, during normal plant operation, some nonessential spaces. The portions of the system applicable to this limiting condition for operation include two separate air handling systems, one comprised of two 100% capacity air handling units AH-25 and the other comprised of two 100% capacity air handling units AH-30 and associated coolers, dampers and ducting. A train consists of one AH-25 supply fan, its associated cooler, one AH-30 exhaust fan, and associated dampers and ducting required to supply air to and from the electrical equipment rooms. Both fans of a train are supplied from the same power source. As the switchgear area coolers are a subsystem of the essential services chilled water systems, it is conservative to use the allowed outage times of Technical Specification 3/4.7.12 (essential services chilled water system) for the switchgear area ventilation system. Technical Specifications are not entered for individual components located in rooms supplied by the switchgear area ventilation system. This is fitting as the air handling units offer complete redundancy (including dampers, coolers and power supplies) and share common ducting to supply and exhaust the affected rooms.

← (DRN 02-1609)

→ (DRN 05-0334, Am. 97)

The provisions of the allowed outage time specified in this TRM may decrease the availability of SVS components. However, the preventive and corrective maintenance intended for these times will improve the reliability of the system. Overall SVS unavailability will be limited by compliance with 10 CFR 50.65 Maintenance Rule Program. This program provides administrative controls to minimize increases in malfunction probability and prevent problems causing excessive SVS outage time.

← (DRN 05-0334, Am. 97)

→ (DRN 02-1609; 04-1191, Am. 91)

If the Switchgear Ventilation System (SVS) is not restored to OPERABLE status within the 72 hour period allowed by the ACTION statement, TRM LCO 3.0.3 must be entered.

← (DRN 02-1609; 04-1191, Am. 91)

3/4.7.13 SWITCHGEAR AREA VENTILATION SYSTEM (cont'd)

→ (DRN 02-1609; 04-1191, Am. 91)

Note the CR evaluation does not relieve any operability requirements for TS systems that may be supported by SVS. The operability of TS equipment must be assured at all times or the appropriate TS action should be entered. Additionally, the design basis functions of SVS should be maintained such that there will not be more than a minimal increase in the likelihood of malfunction, or the consequences of a malfunction, of a SSC important to safety. The primary design basis functions that must be preserved are the ability to maintain a suitable operating environment and the ability to prevent the accumulation of a combustible concentration of hydrogen in the battery rooms.

← (DRN 04-1191, Am. 91)

No additional surveillance requirements are imposed other than those required for the essential services chilled water system.

← (DRN 02-1609)

→ (DRN 02-1876)

3/4.7.14 ESSENTIAL INSTRUMENT AIR

→(EC-935, Am. 117)

Essential Instrument Air (EIA) is designed to provide the motive force for the operation for selected air operated valves that are required for safe shutdown and/or accident mitigation for certain design basis scenarios. Specifically the EIA provides back-up air for CC-641, CC-710, and CC-713 containment isolation air operated valves during post accident operations with a loss of Instrument Air (IA) coincident with a Containment Spray Actuation Signal (CSAS). In addition, it also provides the motive power to containment isolation valves CC-807A, CC-807B, CC-808A, CC-808B, CC-822A, CC-822B, CC-823A, CC-823B, CS-125A, CS-125B and CVC-209 assuming associated Emergency Core Cooling System (ECCS) system malfunctions. In addition, it also provides the motive power to open Shutdown Cooling Suction Isolation Valves SI-405A and SI-405B when necessary with a loss of instrument air (IA).

←(EC-935, Am. 117)

The EIA system is isolated during normal plant operations.

The essential instrument air system consists of 4 stations each with:

- five high pressure accumulator bottles
- high pressure filter
- regulating valve
- relief valve

← (DRN 02-1876)

3/4.7.14 ESSENTIAL INSTRUMENT AIR (cont'd)

The EIA supports the IA system, and is used to supply accumulators/valve operators associated with the above identified safety related containment isolation valves. The EIA is designed to provide air to maintain containment isolation for 30 days post accident. The air in the accumulator stations is designed to be maintained between 2250 and 2500 psig. The system is periodically tested in accordance with the IST program to verify leakdown rates which assures the 30 day capability with an initial pressure of at least 2250 psig.

In some cases, valve specific accumulators do not fully support all required valve actions and manual operation is required to place the EIA system in service. In these cases, a standard time frame of ten (10) hours has been adopted for completion of manual actions. This time period allows simplification of emergency operating procedures related to IA system loss/failure.

The Limiting Condition of Operation is applicable in modes 1, 2, 3 and 4 and it requires all four of the EIA stations to be operable with at least 2250 psig of pressure. The system is designed with two independent sets of air stations (for a total of four individual stations). Each set has a high pressure and low pressure cross-connect. The high pressure cross connect is normally closed and cross connects two stations upstream of the associated station's pressure regulator. Likewise, the low pressure cross connect is normally closed and cross connects two stations downstream of the associated station's pressure regulator. This configuration allows the valves serviced by a particular station to be supported from a backup station if a pressure regulator fails or if pressure is lost in one of the stations.

The actions are designed to accommodate various component level failures and are designed to preserve some level of functionality while efforts are underway to repair the inoperable component(s).

Action A addresses a condition of one station being inoperable (e.g., a failed regulator) and its air bank available (pressure remains $\geq$ 2250 PSIG). The action to align the high pressure and low pressure cross-connect valves within 4 hours ensures full design basis capability to both sets of valves supported by the two stations. The time frame of 4 hours is consistent with the allowed outage time provided in TS 3.6.3 for an inoperable containment isolation valve.

← (DRN 02-1876)

3/4.7.14 ESSENTIAL INSTRUMENT AIR (cont'd)

→ (DRN 04-1191, Am. 91)

Action B addresses a condition with one station inoperable and the design pressure not capable of being made available to its sister station (i.e., pressure <2250 psig or unable to align the HP cross connect). The action for this condition is to align the low pressure cross-connect valves to supply the valves served by the inoperable station within 4 hours. While this action will not necessarily establish full design basis capability for 30 days post accident, it will ensure the design capability for some reduced period of time (i.e., potentially for less than 30 days). An allowed outage time of 14 days is appropriate for this condition as functionality will be provided for some period of time; however, this functionality is not assured for the required 30 days. If operability cannot be restored in 14 days, TRM LCO 3.0.3 must be entered.

← **(DRN 02-1876, 04-1191, Am. 91)**

→(DRN 05-1013, Am. 103)

→(DRN 07-136, Am. 111)

3/4.8.1, 3/4.8.2 A.C. SOURCES

Background

The offsite A.C. circuit voltage requirement is being implemented in response to NRC concerns regarding potentially degraded transmission grids that would not be adequate to support emergency system operation following the trip of a large nuclear unit, the largest operating unit on the grid, or the loss of the most critical transmission line. Of particular concern to the NRC staff is the double sequencing of emergency loads following a reactor trip and subsequent loss-of-offsite-power (LOOP) with the initial start occurring under degraded voltage conditions.

←(DRN 07-136 Am. 111)

A 3-second time delay for a LOOP following a reactor trip in the Steam Generator Tube Rupture (SGTR) analysis was assumed. In support of this analysis, an evaluation of the plant-specific design features was performed that justify the use of the chosen time delay for the consequential LOOP. The following possibilities were addressed that could result in a consequential LOOP: degraded switchyard voltage, spurious switchyard breaker-failure-protection circuit actuation, automatic bus transfer failure, and startup transformer failure.

The evaluation indicated that spurious actuation (i.e., single failure) of switchyard breaker failure protection circuitry, automatic bus transfer failure, or a startup transformer failure could result in a loss of one startup transformer circuit and the corresponding 6.9kV bus (two RCPs) and one 4.16kV safety bus, but would not result in a total LOOP. In addition, because these occurrences would likely be linked to the separation of the main generator from the grid, the half LOOP described would occur with an approximate 7 second time delay from the reactor/turbine trip. The main generator is separated from the grid by a reverse power relay that has been found to actuate at approximately 7 seconds following a turbine trip.

Because the design basis accident (e.g. SGTR) events could involve the actuation of the Emergency Core Cooling Systems (ECCS), the consequences of the delayed LOOP on the performance of the electrical ECCS systems were also evaluated. The consequences of double sequencing and its associated vulnerabilities that would occur as the result of the delayed LOOP were included as a part of this evaluation.

With regard to motor starting capability during a Degraded Voltage / Double Sequencing (DV/DS) scenario, 4kV motor starting capability on offsite power with the 4.16 kV buses at 90 percent bus voltage were evaluated. This value allows for voltage reduction due to SIAS loading and motor starting below the 93.1 percent setting of the degraded voltage relays on the safety buses. The High Pressure Safety Injection (HPSI), Low Pressure Safety Injection (LPSI), Auxiliary Component Cooling Water (ACCW), and emergency feedwater (EFW) pumps could be subject to double sequencing if a LOOP should occur subsequent to the initiation of a Safety Injection Actuation Signal (SIAS). The concern with starting and running motors at degraded voltage conditions is that the motors may overheat and adversely affect qualified life. Based on a number of design margins available

←(DRN 05-1013, Am. 103)

→(DRN 07-136, Am. 111)
3/4.8.1, 3/4.8.2 A.C. SOURCES (Continued)
←(DRN 07-136, Am. 111)

(i.e., minus 10 percent of rated voltage, 10 percent additional thermal capacity for difference in altitude, an increase in thermal capacity due to lower starting ambient temperature, and service factor) on the HPSI, LPSI, ACCW, and EFW 4kV motors, it has been concluded that the increase in motor current for 12.5 seconds (time delay of degraded voltage relay at 93.1 percent) at approximately 90 percent voltage condition, coupled with two consecutive starts (first start at 90 percent bus voltage and second start on the EDG), will not cause any damage to the HPSI, LPSI, ACCW, and EFW motors, nor adversely affect their qualified life.

With regard to 460V AC motor starting capability during a DV/DS scenario, a large number of 460V motors were also identified that may be subject to double sequencing due to cycling on and off by a process signal or SIAS. With the first off-site power start at 90 percent voltage on the 4.16kV buses, the 460V motors will be operating below the 90 percent voltage starting capability specified in motor manufacturers' standard NEMA MG-1. These motors were evaluated and concluded that, based on NEMA MG-1 Sections 12.48 and 12.49 and available design margins, as described in the 4kV motor discussion, it is reasonable to conclude that the increase in motor current for 12.5 seconds due to degraded voltage condition, coupled with two consecutive starts, will not cause any damage to the 460V AC motors.

There is some support in the literature that the concern with starting and running motors at degraded voltage conditions is that the motors may overheat and adversely affect qualified life. NEMA MG-1, Section 20.43.2 states: "it should be recognized that the number of starts should be kept to a minimum since the life of the motor is affected by the number of starts." Institute of Electrical and Electronics Engineers (IEEE) Standard C37.96-2000, IEEE Guide for AC Motor Protection, Section 5.2 .1 states; "It should be noted that deriving increased output at the price of higher temperatures for any given motor means accepting a shorter life." Volume 6, page 6-56 of the EPRI Power Plant Electrical Reference Series states: "Rotor overheating is often far more damaging than stator heating during starting or stalling. Rotors may be weakened so as to fail sooner than they otherwise would have, but no instantaneous melting like that of a fuse element is to be expected under any kind of starting abuse."

Because some motors at Waterford 3 may operate outside their specified requirements during design basis accident DV/DS scenarios (e.g. SGTR), it is necessary for Entergy to be aware of conditions that could place Waterford 3 in such a condition if a design basis accident (e.g. SGTR) were to occur. This can be accomplished through a communication protocol and agreement between Waterford 3 and its transmission system operator (Entergy Transmission) to notify Waterford 3 when a projected trip of Waterford 3 will result in post-trip switchyard voltages that are inadequate to supply emergency loads during a design basis accident (e.g. SGTR) event.
←(DRN 05-1013, Am. 103)

→(DRN 07-136, Am. 111)

3/4.8.1, 3/4.8.2 A.C. SOURCES (Continued)

←(DRN 07-136, Am. 111)

At present, periodic studies are generally performed for Waterford 3 on a 2-3 year interval to project that transmission system voltages will remain stable and within the Waterford 3 operating requirements, under certain contingencies, over the upcoming period. Also daily studies to project post trip voltages are performed for the next day using daily cases representing that day of the month. These studies are updated during the following day if transmission system elements that could affect Waterford 3 post-trip voltages are lost. If a situation is encountered that would result in inadequate switchyard post-trip voltages, the Waterford 3 operating staff will be made aware of the condition.

Additional information regarding double sequencing and grid stability can be found in Section 2.3.6, "Three-Second Time Delay Between Steam Generator Tube Rupture and Loss-of-Offsite Power," of the Safety Evaluation for License Amendment 199 approved and issued by the NRC on April 15, 2005. Reference: regulatory commitment A-26779 in Commitment Management System.

→(DRN 07-136, Am. 111)

The EDG manufacturer's recommended inspection surveillance requirements (SR) were relocated from TS 4.8.1.1.2.f to the TRM. The SR relocation was evaluated against the four criterion of 10 CFR 50.36, Technical Specifications (TS), and determined the SR could be deleted from TS. The performance of the vendor recommended inspections do not impact the ability of the EDG to perform its safety function, therefore, these inspections do not impact the systems capability of mitigating a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Limiting Condition for Operations

The purpose of Offsite A.C. circuit voltage requirement is to ensure that post-trip switchyard voltages will be sufficient to operate emergency loads. Per Nuclear Management Manual Procedure ENS-DC-201, "ENS Transmission Grid Monitoring," the Grid Operator will predict switchyard voltages resulting from a Waterford 3 trip based on actual grid conditions. In accordance with this procedure, the Grid Operator will notify Waterford 3 in the event that the predicted post-trip switchyard voltage reaches a value of 223kV (97% of 230kV). This value provides an alert to control room personnel of grid voltage approaching unacceptable levels. Sufficient margin exists between this value and the minimum acceptable value of approximately 220.8kV (96% of 230kV), which corresponds to the 4kV degraded voltage relay setpoint at 3875V, required to ensure offsite power would remain available for operation of the emergency loads following a design basis accident.

←(DRN 05-1013, 07-136 Am. 111)

➔(DRN 05-1013)

➔ (DRN 07-136, Am. 111)

3/4.8.1, 3/4.8.2 A.C. SOURCES (Continued)

⇐ (DRN 07-136, Am. 111)

Action

When Grid Operator predictions indicate that the Waterford 3 post-trip switchyard voltage will be less than 223kV, Waterford 3 operating staff will be notified. Upon receipt of this information, a condition report should be initiated and an operability assessment performed for the offsite A.C. circuits. The 223kV value will allow time for the Grid Operator to take appropriate action to cope with predicted grid conditions as well as allow time for Waterford 3 to evaluate offsite power operability status. If one or both of the offsite A.C. circuits are determined to be inoperable, then actions should be initiated in accordance with Technical Specification 3.8.1.1.

⇐(DRN 05-1013 Am. 103)

➔ (DRN 07-136 Am. 111)

Surveillance Requirements

The surveillance requirements for the manufacturer's recommended inspections are consistent with the requirements of General Design Criteria (GDC) 17, *Electric Power Systems*, of Appendix A, *General Design Criteria for Nuclear Power Plants*, to 10 CFR 50. These maintenance and inspection activities are included in plant procedures and are performed on a frequency driven and/or condition based approach. Some items are performed on a fixed frequency and others are deferred or modified based on trending and previous inspection results.

⇐(DRN 07-136 Am. 111)

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

→ (DRN 04-1191, Am. 91)

Technical Specification 3/4.8.3 requires that the SUPS be powered from their associated inverters connected to the DC bus. Each of the inverters is normally supplied though its associated rectifier from a 480 V AC MCC. Should this supply fail, the inverters are supplied automatically from a 125 V DC safety related battery. SUPS 3A, 3B and 3AB have a static transfer switch that automatically transfers the load to the bypass transformer if the inverter experiences a predetermined overload, frequency deviation (SUPS 3AB only), undervoltage or overvoltage. In addition, there is a manual transfer switch that allows the operator to connect the load to the bypass transformer if the inverter is not available. Technical Specification 3/4.8.2 requires that the battery and only one of the two battery chargers be operable. When a SUPS rectifier is out of service, the SUPS could be supplied from the DC bus for an indefinite period and still be in compliance with technical specifications. Engineering calculations indicate that battery discharging could occur if a SUPS is being supplied from the DC bus during normal operations (i.e., rectifier out of service) unless the two associated battery chargers are in service. Thus, the potential exists for discharging the battery, even though all applicable technical specifications are being met. If the rectifier is not restored within 24 hours or two battery chargers are not in service within 24 hours, TRM LCO 3.0.3 must be entered.

← (DRN 02-1639; 04-1191, Am. 91)

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The OPERABILITY of the motor-operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

"Containment Penetration Conductor Overcurrent Protection Devices" and "Motor-Operated Valves Thermal Overload Protection and/or Bypass Devices", previously Tables 3.8-1 and 3.8-2, of the Technical Specifications have been incorporated into this manual.

→(LBDCR 13-003, Am. 124)

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

←(LBDCR 13-003, Am. 124)

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

The OPERABILITY of the Fuel Handling Building ventilation system insures that all radioactive material released from an irradiated fuel assembly will be monitored prior to discharge to the atmosphere. The safety analysis for a fuel handling accident in the Fuel Handling Building assumes no filtration and no holdup time.

→ (DRN 02-216)
3/4.11 RADIOACTIVE EFFLUENTS (See note below)
← (DRN 02-216)

BASES

3/4.11.1.1 CONCENTRATION

This Requirement is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than ten 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(e) 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 MPC in air (submersion) was converted to an equivalent concentration in water. This specification does not affect the requirement to comply with the annual limitations of 10 CFR 20.1301(a).

The sampling and analysis of the contents of the regenerative waste tank and the filter flush tank is performed if primary to secondary leakage occurs in a steam generator. The contents of these tanks cannot be discharged to the UNRESTRICTED AREA if any radioactivity is detected in these tanks since the discharge from these tanks is unmonitored. When radioactivity is detected in these tanks, the contents from these tanks must be discharged to the liquid radwaste system where the contents may then be monitored upon discharge.

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 NUREG/CR-4007, Currie, L. A. "Lower Limit Of Detection, Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements" (September 1984).

3/4.11.1.2 DOSE

This Requirement is provided to implement the requirements of Sections II.A, 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.A of Appendix I. The 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 141.16.

→ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.11.1.1 and 3/4.11.1.2 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.
← (DRN 02-216)

3/4.11.1.2 DOSE (Continued)

The dose calculation methodology and parameters implement the requirement 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 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.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the LIQUID RADWASTE TREATMENT SYSTEM ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. 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 requirement implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A 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 fourth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.2.1 DOSE RATE

This requirement provides reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either at or beyond the SITE BOUNDARY in excess of the design objectives of Appendix I to 10 CFR Part 50. This requirement is provided to ensure that gaseous effluents from all units on the site will be appropriately controlled. It provides operational flexibility for releasing gaseous effluents to satisfy the Section II.A and II.C design objectives of Appendix I to 10 CFR Part 50. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for the reduced atmospheric dispersion of gaseous effluents relative to that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM.

→ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.11.1.2, 3/4.11.1.3, and 3/4.11.2.1 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)

3/4.11.2.1 DOSE RATE (Continued)

The specified 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 total body and 3000 mrem/yr to the skin.
These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.
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 NUREG/CR-4007, Currie, L. A. "Lower Limit of Detection, Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements" (September 1984).

3/4.11.2.2 DOSE - NOBLE GASES

The Requirement is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. It implements the guides set forth in Section II.B of Appendix I. The 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 Surveillance 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 dose calculation methodology and parameters established 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.

Grab sampling of effluents from the main condenser evacuation and turbine gland sealing system is not required when this source has been continuously discharging to the plant stack. If no primary to secondary leakage in the steam generator exists, then there should be no radioactive release from the main condenser evacuation and turbine gland sealing system and the gross beta or gamma monitoring for noble gases will be sufficient to determine if any radioactivity is present in the release. If a primary to secondary leak exists, then the release

→ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.11.2.1 and 3/4.11.2.2 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)

3/4.11.2.2 DOSE – NOBLE GASES (Continued)

from the main condenser evacuation and turbine gland sealing systems will be sampled and analyzed in accordance with Table 4.11-2.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This Requirement is provided to implement the requirements of Sections II.C, III.A and IV.KA of Appendix I, 10 CFR Part 50. The Requirements are the guides set forth in Section II.C of Appendix I. The 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 Surveillance 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 requirements for iodine-131, iodine-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 graze with consumption of the milk and meat by man, and (4) Deposition on the ground with subsequent exposure of man.

➞ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.11.2.2 and 3/4.11.2.3 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.
← (DRN 02-216)

3.4.11.2.4 GASEOUS RADWASTE TREATMENT

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 Requirement implements the requirements of 10 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.

3/4.11.3 SOLID RADIOACTIVE WASTE

Solidified wastes are classified in accordance with the requirements of 10 CFR 61.55, as implemented by RW-002-401 2.1.6 and plant waste classification and characterization procedure(s). Annual analysis will be performed on the waste streams to determine the isotopic abundance of gamma emitting isotopes in the streams as described in RW-002-110. Scaling factors for the non-gamma emitting and transuranic constituents will be developed from this annual analysis using RW-002-401 and RW-002-411. The activity of each radionuclide in the solidified waste will be determined by a core sample or a calculational method employing the percent abundance and scaling factors with a dose to curie conversion factor as described in RW-002-401. Solidified wastes will meet the characteristics of 10 CFR 61.56(a). Stabilized wastes will meet the characteristics of 10 CFR 61.56(b). Waste containers will be labeled to identify the waste class. The manifesting requirements of 10 CFR 20.311 are implemented and records are maintained in accordance with 10 CFR 71.91.

3/4.11.4 TOTAL DOSE

The Requirement is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20.1301(d). 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 total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. 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 within a radius of 8 km must be considered. If the dose to any MEMBER OF THE

→ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.11.2.4 and 3/4.11.4 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)

3/4.11.4 TOTAL DOSE (Continued)

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 accordance with the provisions of 40 CFR 190.11 and 10 CFR 2203(a)(4), is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 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 Requirements 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle. Demonstration of compliance with the limit of 40 CFR 190, or with the design objectives of Appendix I to 10 CFR 50 will be considered to demonstrate compliance with the 0.1 Rem limit of 10 CFR 20.1301.

→ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.11.4 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.
← (DRN 02-216)

→ **(DRN 02-216; 02-912)**

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING (See note below)

← **(DRN 02-216; 02-912)**

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this requirement 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 plant 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 was effective for 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 4.12-1 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.

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 NUREG/CR-4007, Currie, L. A. "Lower Limit Of Detection, Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," (September 1984).

→ **(DRN 02-216)**

NOTE: TRM Specification Bases 3/4.12.1 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← **(DRN 02-216)**

➞ (DRN 02-912)

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

← (DRN 02-912)

BASES

➞ (DRN 02-216)

3/4 12.2 LAND USE CENSUS (See note below)

← (DRN 02-216)

This specification 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/yr) 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.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided 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 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.

➞ (DRN 02-216)

NOTE: TRM Specification Bases 3/4.12.2 and 3/4.12.3 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specification Bases requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)