Proposed

Technical Specifications

3.2.B and 4.2.B

Primary Containment Isolation

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

B. Primary Containment Isolation

The primary containment isolation instrumentation for each Trip Function in Table 3.2.2 shall be operable in accordance with Table 3.2.2.

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

B. Primary Containment Isolation

1. The primary containment isolation instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.2.

   When a primary containment isolation channel, and/or the affected primary containment isolation valve, is placed in an inoperable status solely for performance of required instrumentation surveillances, entry into associated Limiting Conditions for Operation and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains isolation capability.

2. Perform a Logic System Functional Test of Primary Containment isolation instrumentation Trip Functions once every Operating Cycle.

Table 3.2.2 (page 1 of 3)
Primary Containment Isolation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	ACTIONS REFERENCED FROM ACTION NOTE 1	TRIP SETTING
1. Main Steam Line Isolation					
a. Low-Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	Note 2.a	≥ 82.5 inches
b. High Main Steam Line Area Temperature	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	8	Note 1	Note 2.a	≤ 196 °F for channels monitoring outside steam tunnel and ≤ 200 °F for channels monitoring inside steam tunnel
c. High Main Steam Line Flow	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2 per main steam line	Note 1	Note 2.a	≤ 140% of rated flow
d. Low Main Steam Line Pressure	RUN	2	Note 1	Note 2.c	≥ 800 psig
e. High Main Steam Line Flow - Not in RUN	STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	Note 2.a	≤ 40% of rated flow
f. Condenser Low Vacuum	RUN, STARTUP/HOT STANDBY(2), HOT SHUTDOWN(2), REFUEL(1 and 2)	2	Note 1	Note 2.a	≤ 12 inches Hg absolute
2. Primary Containment Isolation					
a. Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	Note 2.b	≥ 127.0 inches
b. High Drywell Pressure	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	Note 2.b	≤ 2.5 psig

(1) With reactor coolant temperature > 212 °F.

(2) With any turbine stop valve or turbine bypass valve not closed.

Table 3.2.2 (page 2 of 3)
Primary Containment Isolation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	ACTIONS REFERENCED FROM ACTION NOTE 1	TRIP SETTING
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. High Steam Line Space Temperature	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	6	Note 1	Note 2.d	≤ 196 °F
b. High Steam Line d/p (Steam Line Break)	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	1	Note 1	Note 2.d	≤ 195 inches of water
c. Low Steam Supply Pressure	RUN, STARTUP/HOT STANDBY(3), HOT SHUTDOWN(3), REFUEL(3)	4	Note 1	Note 2.d	≥ 70 psig
d. High Main Steam Line Tunnel Temperature	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	Note 2.d	≤ 200 °F
e. High Main Steam Line Tunnel Temperature Time Delay	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	1	Note 1	Note 2.d	≤ 35 minutes
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. High Main Steam Line Tunnel Temperature	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	Note 2.d	≤ 200 °F
b. High Main Steam Line Tunnel Temperature Time Delay	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	1	Note 1	Note 2.d	≤ 35 minutes
c. High Steam Line Space Temperature	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	6	Note 1	Note 2.d	≤ 196 °F

(1) With reactor coolant temperature > 212 °F.

(3) With reactor steam pressure > 150 psig.

Table 3.2.2 (page 3 of 3)
Primary Containment Isolation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	ACTIONS REFERENCED FROM ACTION NOTE 1	TRIP SETTING
4. RCIC System Isolation (Continued)					
d. High Steam Line d/p (Steam Line Break)	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL$^{(1)}$	1	Note 1	Note 2.d	≤ 195 inches of water
e. High Steam Line d/p Time Delay	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL$^{(1)}$	1	Note 1	Note 2.d	≥ 3 seconds and ≤ 7 seconds
f. Low Steam Supply Pressure	RUN, STARTUP/HOT STANDBY$^{(3)}$, HOT SHUTDOWN$^{(3)}$, REFUEL$^{(3)}$	4	Note 1	Note 2.d	≥ 50 psig
5. Residual Heat Removal Shutdown Cooling Isolation					
a. High Reactor Pressure	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN' REFUEL$^{(1)}$	1	Note 1	Note 2.d	≤ 150 psig

(1) With reactor coolant temperature > 212 °F.

(3) With reactor steam pressure > 150 psig.

Table 3.2.2 ACTION Notes

1. With one or more required Primary Containment Isolation Instrumentation channels inoperable, take all of the applicable Actions in Notes 1.a and 1.b below.

   a. With one or more Trip Functions with one or more required channels inoperable:

      1) For Trip Functions 2.a and 2.b, place inoperable channel in trip within 12 hours; and
      2) For Trip Functions 3.e, 4.b, and 4.e, restore inoperable channel to operable status within 24 hours; and
      3) For all other Trip Functions, place inoperable channel in trip within 24 hours.

   b. With one or more Trip Functions with isolation capability not maintained:

      1) Restore isolation capability within 1 hour.

   Penetration flow paths, isolated as a result of complying with the above Actions, may be unisolated intermittently under administrative controls.

   If any applicable and associated completion time of Note 1.a or 1.b is not met, take the applicable Actions of Note 2 below and referenced in Table 3.2.2 for the channel.

2. a. Isolate the associated Main Steam Line within 12 hours (penetration flow paths may be unisolated intermittently under administrative control); or Place the reactor in HOT SHUTDOWN within 12 hours and place the reactor in COLD SHUTDOWN within the next 12 hours.

   b. Place the reactor in COLD SHUTDOWN within 24 hours.

   c. Place the reactor in STARTUP/HOT STANDBY within 8 hours.

   d. Isolate the affected penetration flow path within 1 hour (penetration flow paths may be unisolated intermittently under administrative control).

Table 4.2.2 (page 1 of 2)
Primary Containment Isolation Instrumentation
Tests and Frequencies

TRIP FUNCTION	CHECK	FUNCTIONAL TEST	CALIBRATION
1. Main Steam Line Isolation			
a. Low-Low Reactor Vessel Water Level	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
b. High Main Steam Line Area Temperature	NA	Every 3 Months	Each Refueling Outage
c. High Main Steam Line Flow	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
d. Low Main Steam Line Pressure	NA	Every 3 Months	Every 3 Months
e. High Main Steam Line Flow - Not in RUN	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
f. Condenser Low Vacuum	NA	Every 3 Months	Every 3 Months
2. Primary Containment Isolation			
a. Low Reactor Vessel Water Level	NA	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
b. High Drywell Pressure	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
3. High Pressure Coolant Injection (HPCI) System Isolation			
a. High Steam Line Space Temperature	NA	Every 3 Months	Each Refueling Outage
b. High Steam Line d/p (Steam Line Break)	NA	Every 3 Months	Every 3 Months
c. Low Steam Supply Pressure	NA	Every 3 Months	Every 3 Months
d. High Main Steam Line Tunnel Temperature	NA	Every 3 Months	Each Refueling Outage
e. High Main Steam Line Tunnel Temperature Time Delay	NA	NA	Once/Operating Cycle

(1) Trip unit calibration only.

Table 4.2.2 (page 2 of 2)
Primary Containment Isolation Instrumentation
Tests and Frequencies

TRIP FUNCTION	CHECK	FUNCTIONAL TEST	CALIBRATION
4. Reactor Core Isolation Cooling (RCIC) System Isolation			
a. High Main Steam Line Tunnel Temperature	NA	Every 3 Months	Each Refueling Outage
b. High Main Steam Line Tunnel Temperature Time Delay	NA	NA	Once/Operating Cycle
c. High Steam Line Space Temperature	NA	Every 3 Months	Each Refueling Outage
d. High Steam Line d/p (Steam Line Break)	NA	Every 3 Months	Every 3 Months
e. High Steam Line d/p (Steam Line Break) Time Delay	NA	Every 3 Months	Every 3 Months
f. Low Steam Supply Pressure	NA	Every 3 Months	Every 3 Months
5. Residual Heat Removal Shutdown Cooling Isolation			
a. High Reactor Pressure	NA	Every 3 Months	Every 3 Months

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Bases

3.2.B and 4.2.B

Primary Containment Isolation

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

BACKGROUND

The primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary 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 DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and reactor coolant pressure boundary (RCPB) isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient temperatures, (c) main steam line (MSL) flow, (d) main steam line pressure, (e) condenser vacuum, (f) drywell pressure, (g) high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) steam line d/p, (h) HPCI and RCIC steam line pressure, and (i) reactor vessel pressure. Redundant sensor input signals from each parameter are provided for initiation of isolation.

Primary containment isolation instrumentation has inputs to the trip logic of the isolation functions listed below.

1. Main Steam Line Isolation

The Low - Low Reactor Vessel Water Level, Low Main Steam Line Pressure, High Main Steam Line Flow - Not in RUN, and Condenser Low Vacuum Trip Functions each receive inputs from four channels. The outputs of these channels are combined in a one-out-of-two taken twice logic to initiate isolation of all main steam isolation valves (MSIVs), MSL drain valves, and recirculation loop sample isolation valves.

The High Main Steam Line Flow Trip Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to cause an isolation of all MSIVs, MSL drain valves, and recirculation sample isolation valves. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings are arranged in a one-out-of-two taken twice logic. This is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation.

The High Main Steam Line Area Temperature Trip Function receives input from 16 channels, four for each of four main steam line areas. The logic is arranged similar to the High Main Steam Line Flow Trip Function. One channel from each steam tunnel area inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause isolation.

MSL Isolation Trip Functions isolate the Group 1 valves.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

BACKGROUND (continued)

2. Primary Containment Isolation

The Low Reactor Vessel Water Level and High Drywell Pressure Trip Functions each receive inputs from four channels. For each Trip Function, the outputs of these channels are combined in a one-out-of-two taken twice logic to initiate isolation of the PCIVs identified in Reference 1.

Primary Containment Isolation Trip Functions isolate the Groups 2, 3, and 4 valves. Group 5 valves are also isolated by the Low Reactor Vessel Water Level Trip Function.

3, 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

The HPCI High Steam Line d/p, RCIC High Steam Line d/p, and RCIC High Steam Line d/p Time Delay Trip Functions each receive input from two channels, with each channel in one trip system using a one-out-of-one logic. The trip systems are arranged in a one-out-of-two logic. Each of the two trip systems is connected to both valves on the associated penetration.

The HPCI and RCIC Low Steam Supply Line Pressure Trip Functions each receive input from four steam supply pressure channels. The outputs from the associated steam supply pressure channels are connected in a one-out-of-two-twice trip system logic arrangement. There are two trip system logics which provide input to one trip system. The trip system must trip to initiate isolation of both valves on the associated penetration.

The HPCI and RCIC High Main Steam Line Tunnel Temperature Trip Functions each receive input from 4 channels. Four channels, each with an associated temperature switch, are connected in a one-out-of-two-twice arrangement which provides input to two trip systems. Both trip systems must trip to initiate isolation of both valves on the associated penetration. In addition, the HPCI and RCIC High Main Steam Line Tunnel Temperature Trip Functions each have time delays. These Time Delay Trip Functions each receive input from two channels, with each channel in one of the trip system using a one-out-of-one logic. The trip systems are arranged in a one-out-of-two logic.

The HPCI and RCIC High Steam Line Space Temperature Trip Functions each receive input from 12 channels. There are three steam line areas each monitored by one set of four channels. One channel from each of the three steam line areas inputs to one of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to cause an isolation of both valves on the associated penetration. The trip strings are arranged in a one-out-of-two taken twice logic. This is effectively a one-out-of-six taken twice logic arrangement to initiate isolation.

HPCI System and RCIC System Isolation Trip Functions isolate the Group 6 valves, as appropriate.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

BACKGROUND (continued)

5. Residual Heat Removal Shutdown Cooling Isolation

The High Reactor Pressure Trip Function receives input from two channels. The outputs from these channels are arranged in a one-out-of-two logic to initiate isolation of the Shutdown Cooling (SDC) supply isolation valves.

The Residual Heat Removal Shutdown Cooling Isolation Trip Function isolates the Group 4 SDC supply isolation valves.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of Reference 2 to initiate closure of valves to limit offsite doses.

Primary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Trip Functions are retained for other reasons and are described below in the individual Trip Functions discussion.

The operability of the primary containment isolation instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions specified in Table 3.2.2. Each Trip Function must have the required number of operable channels in each trip system, with their trip setpoints within the calculational as-found tolerances specified in plant procedures. Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.2. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Certain Emergency Core Cooling Systems (ECCS) valves (e.g., containment spray isolation valves) also serve the dual function of automatic PCIVs. The signals that isolate these valves are also associated with the automatic initiation of the ECCS. Some instrumentation requirements and Actions associated with these signals are addressed in Specification 3.2.A, "Emergency Core Cooling Systems (ECCS)," and are not included in this specification.

In general, the individual Trip Functions are required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, and REFUEL (with reactor coolant temperature > 212°F) consistent with the Applicability for Primary Containment Integrity requirements in Specification 3.7.A.2. Trip Functions that have different Applicabilities are discussed below in the individual Trip Functions discussion.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Trip Function by Trip Function basis.

Main Steam Line Isolation

1.a. Low - Low Reactor Vessel Water Level

Low reactor pressure vessel (RPV) water level indicates that the capability to

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Low - Low Reactor Vessel Water Level Trip Function is one of the many Trip Functions assumed to be operable and capable of providing isolation signals. The Low - Low Reactor Vessel Water Level Trip Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 3). The isolation of the MSLs supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Low - Low Reactor Vessel Water Level Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Low - Low Reactor Vessel Water Level Trip Setting is chosen to be the same as the ECCS Low - Low Reactor Vessel Water Level Trip Setting (Specification 3.2.A) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits. The Trip Setting is referenced from the top of enriched fuel.

This Function isolates the Group 1 valves.

1.b. High Main Steam Line Area Temperature

Main steam line tunnel temperature is provided to detect a leak in the RCPB in the steam tunnel and provides diversity to the high flow instrumentation. Temperature is sensed in four different areas of the steam tunnel in the vicinity of the main steam lines. The isolation occurs when a very small leak has occurred in any one of the four areas. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Main steam line area temperature signals are initiated from a total of sixteen temperature switches located in the four areas being monitored. Sixteen channels of High Main Steam Line Area Temperature Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The High Main Steam Line Area Temperature Trip Setting is chosen to provide early indication of a steam line break.

These Functions isolate the Group 1 valves.

1.c. High Main Steam Line Flow

High Main Steam Line Flow is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore,

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the isolation is initiated on high flow to prevent or minimize core damage. The High Main Steam Line Flow Trip Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 4). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

The MSL flow signals are initiated from 16 differential pressure transmitters that are connected to the four MSLs (the differential pressure transmitters sense differential pressure across a flow restrictor). The differential pressure transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of High Main Steam Line Flow Trip Function for each MSL (two channels per trip system) are available and are required to be operable so that no single instrument failure will preclude detecting a break in any individual MSL.

The Trip Setting is chosen to ensure that fuel peak cladding temperature and offsite dose limits are not exceeded due to the break.

This Trip Function isolates the Group 1 valves.

1.d. Low Main Steam Line Pressure

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Low Main Steam Line Pressure Trip Function is directly assumed in the analysis of the pressure regulator failure (Ref. 5). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Trip Function supports actions to ensure that Safety Limit 1.1.B is not exceeded. (This Trip Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RATED THERMAL POWER.)

The MSL low pressure signals are initiated from four pressure switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each pressure switch is able to detect low MSL pressure. Four channels of Low Main Steam Line Pressure Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Setting was selected to be high enough to prevent excessive RPV depressurization.

The Low Main Steam Line Pressure Trip Function is only required to be operable in the RUN Mode since this is when the assumed transient can occur (Ref. 5).

This Trip Function isolates the Group 1 valves.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e. High Main Steam Line Flow - Not in RUN

High Main Steam Line Flow when the reactor mode switch is not in RUN provides protection for a turbine pressure regulator malfunction which causes the turbine control valves and turbine bypass valves to open or protection for a main steam line break. These events would result in a rapid depressurization and cooldown of the RPV. The High Main Steam Line Flow - Not in RUN Trip Function was credited in the MSLB at low power analysis.

The MSL flow signals are initiated from 4 differential pressure transmitters, one connected to each of the four MSLs (the differential pressure switches sense differential pressure across a flow restrictor). Four channels of High Main Steam Line Flow - Not in RUN Trip Function (two channels per trip system) are available and are required to be operable so that no single instrument failure will preclude providing protection against a turbine pressure regulator malfunction or a break in any individual MSL.

The Trip Setting is chosen to provide early indication of a steam line break.

The High Main Steam Line Flow - Not in RUN Trip Function is only required to be operable in the STARTUP/HOT STANDBY, HOT SHUTDOWN, and REFUEL (with reactor coolant temperature > 212°F) Modes. In the RUN Mode, protection for the depressurization resulting from a turbine pressure regulator malfunction is provided by the Low Main Steam Line Pressure Trip Function and protection for depressurization resulting from a main steam line break is provided by the High Main Steam Line Flow Trip Function.

This Trip Function isolates the Group 1 valves.

1.f. Low Condenser Vacuum

The Low Condenser Vacuum Trip Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Low Condenser Vacuum Trip Function is assumed to be operable and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure switches that sense the pressure in the condenser. Four channels of Low Condenser Vacuum Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Setting is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As indicated in Footnote (2) to Table 3.2.2, the channels are not required to be operable in the STARTUP/HOT STANDBY, HOT SHUTDOWN, and REFUEL (with reactor coolant temperature > 212°F) Modes when all turbine stop valves (TSVs) and turbine bypass valves (TBVs) are closed, since the potential for condenser overpressurization is minimized. A key lock switch is provided to manually bypass the Low Condenser Vacuum Trip Function channels to enable plant

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

startup and shutdown when condenser vacuum is greater than 12 inches Hg absolute and all TSVs and TBVs are closed.

This Trip Function isolates the Group 1 valves

Primary Containment Isolation

2.a. Low Reactor Vessel Water Level

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on low RPV water level supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Low Reactor Vessel Water Level Trip Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Low Reactor Vessel Water Level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Low Reactor Vessel Water Level Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Low Reactor Vessel Water Level Trip Setting was chosen to be the same as the RPS Low Reactor Vessel Water Level scram Trip Setting (Specification 3.1.A), since isolation of these valves is not critical to orderly plant shutdown. The Trip Setting is referenced from the top of enriched fuel.

This Trip Function isolates the Groups 2, 3, 4, and 5 valves.

2.b. High Drywell Pressure

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The High Drywell Pressure Trip Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of High Drywell Pressure Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Setting was selected to be the same as the ECCS High Drywell Pressure (Specification 3.2.A) and RPS High Drywell Pressure (Specification 3.1.A) Trip Settings, since this may be indicative of a LOCA inside primary containment.

This Trip Function isolates the Groups 2, 3 and 4 valves.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

3.a, 4.c HPCI and RCIC High Steam Line Space Temperature

High Steam Line Space Temperature Trip Functions are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Trip Functions are not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

High Steam Line Space Temperature signals are initiated from temperature switches that are appropriately located to detect a leak from the system piping that is being monitored. For each Trip Function, there are four instruments that monitor each of three locations. Twelve channels for HPCI High Steam Line Space Temperature are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function. Twelve channels for RCIC High Steam Line Space Temperature are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Settings are set high enough above anticipated normal operating levels to avoid spurious isolation, yet low enough to provide timely detection of a HPCI or RCIC steam line break.

These Trip Functions isolate the associated Group 6 valves.

3.b., 4.d. HPCI and RCIC High Steam Line d/p (Steam Line Break)

High Steam Line d/p (Steam Line Break) Trip Functions are provided to detect a break of the RCIC or HPCI steam lines and initiate closure of the steam line isolation valves of the appropriate system. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and the core can uncover. Therefore, the isolations are initiated on high d/p to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the requirements of 10 CFR 50.46 are met. Specific credit for these Trip Functions is not assumed in any UFSAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

The HPCI and RCIC High Steam Line d/p (Steam Line Break) signals are initiated from differential pressure switches (two for HPCI and two for RCIC) that are connected to the associated system steam lines. Two channels of both HPCI and RCIC High Steam Line d/p (Steam Line Break) Trip Functions are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Trip Settings are set high enough above anticipated normal operating levels to avoid spurious isolation, yet low enough to provide timely detection of a HPCI or RCIC steam line break.

These Trip Functions isolate the associated Group 6 valves.

3.c., 4.f. HPCI and RCIC Low Steam Supply Pressure

Low steam supply pressure indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system's turbine. These isolations are for equipment protection. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications because of the potential for possible system initiation failure resulting from these instruments.

The HPCI and RCIC Low Steam Supply Pressure signals are initiated from pressure switches (four for HPCI and four for RCIC) that are connected to the associated system steam line. Four channels of both HPCI and RCIC Low Steam Supply Pressure Trip Functions are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Settings are selected to be below the pressure at which the system's turbine can effectively operate.

Since these Trip Functions are provided for equipment protection, they are only required to be operable when the HPCI and RCIC System are required to be operable. Therefore, as indicated in Footnote (3) to Table 3.2.2, in the STARTUP/HOT STANDBY, HOT SHUTDOWN, and REFUEL Modes, the channels are only required to be operable when reactor steam pressure is > 150 psig.

These Trip Functions isolate the associated Group 6 valves.

3.d., 3.e., 4.a., 4.b. HPCI and RCIC High Main Steam Line Tunnel Temperature and Time Delay

HPCI and RCIC High Main Steam Line Tunnel Temperature Trip Functions are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Trip Functions are not assumed in any UFSAR transient or accident analysis., since bounding analyses are performed for large breaks such as recirculation or MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

HPCI and RCIC High Main Steam Line Tunnel Temperature signals are initiated from temperature switches that are appropriately located to detect a leak from the associated system piping that is being monitored. For each Trip Function, there are four instruments that monitor the area. Four channels for HPCI High Main Steam Line Tunnel Temperature are available and are

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

required to be operable to ensure that no single instrument failure can preclude the isolation function. Four channels for RCIC High Main Steam Line Tunnel Temperature are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Settings are set high enough above anticipated normal operating levels to avoid spurious isolation, yet low enough to provide timely detection of a HPCI or RCIC steam line break.

These Trip Functions isolate the associated Group 6 valves.

4.e RCIC High Steam Line d/p Time Delay

The RCIC High Steam Line d/p Time Delay is provided to prevent false isolations on RCIC High Steam Line d/p during system startup transients and therefore improves system reliability. This Trip Function is not assumed in any UFSAR transient or accident analyses.

The RCIC High Steam Line d/p Time Delay Trip Function delays the RCIC High Steam Line d/p (Steam Line Break) signal by use of time delay relays. When a RCIC High Steam Line d/p (Steam Line Break) signal is generated, the time delay relays delay the tripping of the associated RCIC isolation trip system for a short time. Two channels of RCIC High Steam Line d/p Time Delay Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

The Trip Setting is chosen to be long enough to prevent false isolations due to system starts but not so long as to impact compliance with 10CFR50.46 requirements.

This Trip Function, in conjunction with the RCIC High Steam Line d/p (Steam Line Break) Trip Function, isolates the RCIC System Group 6 valves.

Residual Heat Removal Shutdown Cooling Isolation

5.a. High Reactor Pressure

The High Reactor Pressure Trip Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

The High Reactor Pressure signals are initiated from two pressure switches. Two channels of High Reactor Pressure Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function. The Trip Function is only required to be operable in the RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, and REFUEL (with reactor coolant temperature > 212°F) Modes, since these are the only Modes in which the reactor can be pressurized; thus, equipment protection is needed.

The Trip Setting was chosen to be low enough to protect the system equipment from overpressurization.

This Trip Function isolates the Group 4 SDC supply isolation valves.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

ACTIONS

Table 3.2.2 ACTION Note 1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Trip Function (12 hours for those Trip Functions that have channel components common to RPS instrumentation, i.e., Trip Functions 2.a and 2.b, and 24 hours for those Trip Functions that do not have channel components common to RPS instrumentation, i.e., all other Trip Functions), has been shown to be acceptable (Refs. 6 and 7) to permit restoration of any inoperable channel to operable status. This out of service time is only acceptable provided the associated Trip Function is still maintaining isolation capability (refer to the next paragraph). For all Trip Functions except for Trip Functions 3.e, 4.b, and 4.e, if the inoperable channel cannot be restored to operable status within the allowable out of service time, the channel must be placed in the tripped condition per Table 3.2.2 ACTION Note 1.a.1) or 1.a.3), as applicable. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), the applicable actions of Table 3.2.2 ACTION Note 2 must be taken. For Trip Functions 3.e, 4.b, and 4.e, Table 3.2.2 ACTION Note 1.a.2) requires the channel to be restored to operable status. Table 3.2.2 ACTION Note 1.a.2) does not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

Table 3.2.2 ACTION Note 1.b is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Trip Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The Trip Functions are considered to be maintaining isolation capability when sufficient channels are operable or in trip, such that both trip systems will generate a trip signal from the given Trip Function on a valid signal. For Trip Functions 1.a, 1.d, 1.e, 1.f, 2.a, 2.b, 3.b, 3.d, 4.a, 4.d, and 5.a, this would require both trip systems to have one channel operable or in trip. For Trip Function 1.c, this would require both trip systems to have one channel, associated with each MSL, operable or in trip. Trip Functions 1.b, 3.a and 4.c, consist of channels that monitor several locations within a given area (e.g., different locations within the main steam tunnel area). Therefore, this would require both trip systems to have one channel per location operable or in trip. For Trip Functions 3.e, 4.b and 4.e, this would require both trip systems to have one channel operable. For Trip Functions 3.c and 4.f (which only have one trip system for each Trip Function), this would require one trip system to have one channel in each trip system logic operable or in trip.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

ACTIONS (continued)

Table 3.2.2 ACTION Note 1 also allows penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

Table 3.2.2 ACTION Notes 2.a, 2.b, 2.c and 2.d

If any applicable Action and associated completion time of Table 3.2.2 ACTION Note 1.a or 1.b are not met, the applicable Actions of Table 3.2.2 ACTION Note 2 and referenced in Table 3.2.2 (as identified for each Trip Function in the Table 3.2.2 "ACTIONS REFERENCED FROM ACTION NOTE 1" column) must be immediately entered and taken. The applicable Action specified in Table 3.2.2 is Trip Function and Mode or other specified condition dependent.

For Table 3.2.2 ACTION Note 2.a, if the channel is not restored to operable status or placed in trip within the allowed Completion Time the associated MSLs may be isolated, and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. This action will generally only be used if a Trip Function 1.c channel is inoperable and untripped. The associated MSL(s) to be isolated are those whose High Main Steam Line Flow Trip Function channel(s) are inoperable. Alternately, the plant must be placed in a Mode or other specified condition in which the LCO does not apply. This is done by placing the plant in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 12 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Table 3.2.2 ACTION Note 2.a also allows penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

For Table 3.2.2 ACTION Note 2.b, if the channel is not restored to operable status or placed in trip within the allowed Completion Time, the plant must be placed in a Mode or other specified condition in which the LCO does not apply. This is done by placing the plant in COLD SHUTDOWN within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

For Table 3.2.2 ACTION Note 2.c, if the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a Mode or other specified condition in which the LCO does not apply. This is done by placing the plant in at least STARTUP/HOT STANDBY within 8 hours. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach STARTUP/HOT STANDBY from full power conditions in an orderly manner and without challenging plant systems.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

ACTIONS (continued)

For Table 3.2.2 ACTION Note 2.d, if the channel is not restored to operable status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channel. The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s). Table 3.2.2 ACTION Note 2.d also allows penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.B.1

As indicated in Surveillance Requirement 4.2.B.1, primary containment isolation instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.2. Table 4.2.2 identifies, for each Trip Function, the applicable Surveillance Requirements.

Surveillance Requirement 4.2.B.1 also indicates that when a channel (and/or the affected PCIV) is placed in an inoperable status solely for performance of required instrumentation Surveillances, entry into associated LCO and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable LCO entered and required Actions taken. This allowance is based on the reliability analysis (Refs. 6 and 7) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary.

Surveillance Requirement 4.2.B.2

The Logic System Functional Test demonstrates the operability of the required initiation logic and simulated automatic operation for a specific channel. The automatic initiation testing required by the PCIV Technical Specifications overlaps this Surveillance to provide testing of the assumed safety function. For Main Steam Line Isolation Trip Functions, a simulated automatic actuation, which opens all pilot valves of the main steam line isolation valves, shall be performed such that each trip system logic can be verified independent of its redundant counterpart. The Frequency of "once every Operating Cycle" is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the specified Frequency.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

SURVEILLANCE REQUIREMENTS (continued)

Table 4.2.2, Check

Performance of an Instrument Check once per day for Trip Functions 1.a, 1.b, 1.e, 1.h, and 2.b, ensures that a gross failure of instrumentation has not occurred. An Instrument Check is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. An Instrument Check will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each Calibration. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit. The Frequency is based upon operating experience that demonstrates channel failure is rare. The Instrument Check supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

Table 4.2.2, Functional Test

For Trip Functions 1.a, 1.b, 1.c, 1.d, 1.e, 1.f, 2.a, 2.b, 3.a, 3.b, 3.c, 3.d, 4.a, 4.c, 4.d, 4.e, 4.f, and 5.a, a Functional Test is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of "Every 3 Months" is based on the reliability analysis of References 6 and 7.

Table 4.2.2, Calibration

For Trip Functions 1.a, 1.b, 1.c, 1.d, 1.e, 1.f, 2.a, 2.b, 3.a, 3.b, 3.c, 3.d, 3.e, 4.a, 4.b, 4.c, 4.d, 4.e, 4.f, and 5.a, an Instrument Calibration is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. An Instrument Calibration leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The specified Instrument Calibration Frequencies are based upon the time interval assumptions for calibration used in the determination of the magnitude of equipment drift in the associated setpoint analyses.

BASES: 3.2.B/4.2.B PRIMARY CONTAINMENT ISOLATION

SURVEILLANCE REQUIREMENTS (continued)

For Trip Functions 1.a, 1.c, 1.e, 2.a, and 2.b, a calibration of the trip units is required (Footnote (1)) once every 3 months. Calibration of the trip units provides a check of the actual setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the calculational as-found tolerances specified in plant procedures. The Frequency of every 3 months is based on the reliability analysis of References 6 and 7 and the time interval assumption for trip unit calibration used in the associated setpoint calculation.

REFERENCES

1. Technical Requirements Manual.

2. UFSAR, Chapter 14.

3. UFSAR, Table 6.5.3.

4. UFSAR, Section 14.6.5.

5. UFSAR, Section 14.5.4.1.

6. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

7. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

Current

Technical Specifications

Markups

3.2.B and 4.2.B

Primary Containment Isolation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the operational status of the plant instrumentation systems which initiate and control a protective function.

Objective:

To assure the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

When the system(s) it initiates or controls is required in accordance with Specification 3.5, the instrumentation which initiates the emergency core cooling system(s) shall be operable in accordance with Table 3.2.1.

B. Primary Containment Isolation

A.2 L.1

When primary containment integrity is required, in accordance with Specification 3.7, ~~the instrumentation that initiates primary containment isolation~~ shall be operable in accordance with Table 3.2.2.

The primary containment isolation instrumentation for each Trip Function in Table 3.2.2

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

The instrumentation that initiates the isolation of the reactor building ventilation system and the actuation of the standby gas treatment system shall be operable in accordance with Table 3.2.3.

primary containment isolation

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the surveillance requirements of the instrumentation systems which initiate and control a protective function.

Objective:

To verify the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.1.

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1. The primary containment isolation

B. Primary Containment Isolation

A.3 A.4

~~Instrumentation and logic systems~~ shall be ~~functionally tested and calibrated~~ checked, as indicated in Table 4.2.2.

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C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.3.

When a channel, and/or the affected primary containment isolation valve, is placed in an inoperable status solely for performance of required instrumentation surveillances, entry into associated Limiting Conditions for Operation may be delayed for up to 6 hours provided the associated Trip Function maintains isolation capability.

A.5

A.1

PRIMARY CONTAINMENT ISOLATION — A.7

TABLE 3.2.~~1~~ 2
(Cont'd)

~~EMERGENCY CORE COOLING SYSTEM ACTUATION~~ INSTRUMENTATION

~~Low Pressure Coolant Injection System A & B (Note 1)~~

A.8 — Actions when Required Channels are Inoperable — Actions Referenced from Action Note 1

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip ~~Level~~ Setting (A.1)	~~Required ACTION When Minimum Conditions For Operation Are Not Satisfied~~
1 (Note 8)	Low Reactor Pressure (PT-2-3-56C/D(M))	300 ≤ p ≤ 350 psig	Note 11
2 (Note 8)	High Drywell Pressure (PT-10-101(A-D)(M))	≤2.5 psig	Note 10
2 (Note 8)	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(S1))	≥82.5" above top of enriched fuel	Note 10
1 (Note 9)	Reactor Vessel Shroud Level (LT-2-3-73A/B(M))	≥2/3 core height	Note 13
1 (Note 9)	Time Delay (10A-K72A & B)	≤60 seconds	Note 12
1 (Note 9)	Pump Start Time Delay (10A-K50A & B)	3 ≤ t ≤ 5 seconds	Note 12
5.a 1 ~~(Note 9)~~ A.5	~~Low~~ High Reactor Pressure A.6 ~~(PS-2-128A & B)~~ LA.1	LA.2 ~~100 ≤~~ p ≤ 150 psig	Note 1 ~~Note 10~~ 2.d A.8
2 per pump (Note 8)	RHR Pump (A-D) Discharge Pressure (PS-10-105(A-H))	≥100 psig	Note 18
2 (Note 8)	High Drywell Pressure (PT-10-101(A-D)(S1))	≤2.5 psig	Note 13

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A.9

A.1

PRIMARY CONTAINMENT ISOLATION

A.7

A.8

TABLE 3.2.~~1~~ 2
(Cont'd)

Actions when Required Channels are Inoperable

Actions Referenced from Action Note 1

~~EMERGENCY CORE COOLING SYSTEM ACTUATION~~ INSTRUMENTATION

~~Low Pressure Coolant Injection System A & B (Note 1)~~

A.1

Minimum Number of Operable Instrument Channels per Trip System		Trip Function	Trip ~~Level~~ Setting	~~Required ACTION When Minimum Conditions For Operation Are Not Satisfied~~
1	(Note 9)	Time Delay (10A-K45A & B)	$\leq$6 minutes	Note 12
2	(Note 8)	Low Reactor Pressure (PT-2-3-56A/B(M) & PT-2-3-52C/D(M))	300 $\leq$ p $\leq$ 350 psig	Note 11
1	(Note 8)	Auxiliary Power Monitor (LNPX C/D)	--	Note 10
1	(Note 8)	Pump Bus Power Monitor (27/3A/B, 27/4A/B)	--	Note 10
5.b ~~1~~		~~Trip System Logic~~	~~--~~	~~Note 5~~

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2 ACTION

TABLE 3.2.~~7~~ NOTES

1. Each of the two Core Spray, LPCI and RPT, subsystems are initiated and controlled by a trip system. The subsystem "B" is identical to the subsystem "A".

2. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits. If the channel cannot be tripped by the means stated above, that channel shall be made operable within 24 hours or an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

3. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.

4. One trip system with initiating instrumentation arranged in a one-out-of-two logic.

A.9

5. ~~If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.~~ L.4

6. Any one of the two trip systems will initiate ADS. If the minimum number of operable channels in one trip system is not available, the requirements of Specification 3.5.F.2 and 3.5.F.3 shall apply. If the minimum number of operable channels is not available in both trip systems, Specifications 3.5.F.3 shall apply.

7. One trip system arranged in a two-out-of-two logic.

8. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions For Operation and required ACTIONS may be delayed for up to 6 hours provided the associated Trip Function or redundant Trip Function maintains ECCS initiation capability or Recirculation Pump Trip capability.

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9. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions For Operation and required ACTIONS may be delayed for up to 6 hours. A.5

Action Note 1

10. With one or more channels inoperable ~~for Core Spray and/or LPCI~~:

Action Note 1.b

~~A~~ Within one hour from discovery of loss of ~~initiation~~ Isolation capability ~~for feature(s) in one division~~, ~~declare the associated systems inoperable~~, and A.7 L.2 M.1

Action Note 1.a.3)

~~B~~ Within 24 hours, place channel in trip.

Action Notes 1 & 2.d

~~C~~ If required actions and associated completion times of actions A or B are not met, ~~immediately declare the associated systems inoperable~~. M.1

11. With one or more channels inoperable for injection permissive and/or recirculation discharge valve permissive:

A. Within one hour from discovery of loss of initiation capability for feature(s) in one division, declare the associated systems inoperable, and

B. Within 24 hours, restore channel to operable status.

C. If required actions and associated completion times of actions A or B are not met, immediately declare the associated systems inoperable.

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VYNPS

TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

ACTIONS REFERENCED FROM ACTION NOTE 1

	Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required ACTION When Minimum Conditions For Operation Are Not Satisfied (Note 2)
1.a	2 (Notes 11, 12) A.5 A.11	Low-Low Reactor Vessel Water Level (LT-2-3-57A/B(S2), LT-2-3-58A/B(S2))	>82.5" above the top of enriched fuel LA.3	A NOTE 2.a L.3
1.b	M.2 8 2 of 4 in each of 2 channels (Notes 11, 12)	High Main Steam Line Area Temperature (TS-2-(121-124)(A-D))	<212°F ≤196°F FOR CHANNELS MONITORING OUTSIDE STEAM TUNNEL AND ≤200°F FOR CHANNELS MONITORING INSIDE STEAM TUNNEL M.7	B NOTE 2.a M.3
1.c	A.5 2/steam line (Notes 11, 12)	High Main Steam Line Flow (DPT 2-(116-119)(A-D)(M))	<140% of rated flow	B NOTE 2.a M.3
1.d	2 (Notes 1, 11, 12) A.13	Low Main Steam Line Pressure (PS-2-134(A-D)) LA.1	>800 psig	B NOTE 2.c
1.e	2 (Notes 6, 11, 12) A.14	High Main Steam Line Flow (DPT-2-116A, 117B, 118C, 119D(S1)) - NOT IN RUN A.12	<40% of rated flow	B NOTE 2.a M.3
2.a	A.5 2 (Notes 11, 12)	Low Reactor Vessel Water Level (LT-2-3-57A/B(M), LT-2-3-58A/B(M))	Same as Reactor Protection System ≥127.0 INCHES A.1	A NOTE 2.b
2.b	A.5 2 (Notes 11, 12)	High Drywell Pressure	Same as Reactor Protection System ≤2.5 PSIG A.1	A NOTE 2.b
1.f	2 (Notes 10, 11, 12)	Condenser Low Vacuum	<12" Hg absolute	A NOTE 2.a L.3
	A.15 1	Trip System Logic	--	A L.4

A.1

TABLE 3.2.2
(Cont'd)

Actions Referenced from Action Note 1

HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

	Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip ~~Level~~ Setting A.1	~~Required ACTION When Minimum Conditions For Operation Are Not Satisfied~~
3.a	~~2 per set of 4~~ 6 A.16 (Notes 11, 12) A.5 A.11	High Steam Line Space Temperature (TS-23-101-104)(B-D) LA.1	~~≤212°F~~ ≤ 196°F M.7	Note 3 2.d
3.b	1 (Notes 11, 12) A.5 A.11	High Steam Line d/p (Steam Line Break) (DPIS-23-76/77)	≤195 inches of water	Note 3 2.d
3.c	4 (Notes 5, 11, 13) A.5 A.11 LA.7	Low HPCI Steam Supply Pressure (PS-23-68 (A-D))	≥70 psig	Note 3 2.d
3.d	2 (Notes 11, 12) A.5 A.11	HIGH Main Steam Line Tunnel Temperature (TS-23-(101-104)A)	~~≤212°F~~ ≤ 200°F M.7	Note 3 2.d
3.e	1 (Notes 11, 12) A.5 A.11	Time Delay (23A-K48) (23A-K49)	≤35 minutes	Note 3 2.d
	~~1~~	~~Trip System Logic~~		~~Note 3~~ L.4

HIGH MAIN STEAM TUNNEL TEMPERATURE

A.1

TABLE 3.2.2
(Cont'd)

REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

Actions Referenced from Action Note 1

	Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip ~~Level~~ Setting (A.1)	Required ACTION When Minimum Conditions For Operation Are Not Satisfied
4.a	2 (Notes ~~11~~, ~~12~~) A.5 A.11	HIGH Main Steam Line Tunnel Temperature ~~(TS-13-(79-82)A)~~	~~≤212°F~~ ≤ 200 °F M.7	Note ~~3~~ 2.d
4.b	1 (Notes ~~11~~, ~~12~~) A.5 A.11	HIGH MAIN STEAM LINE TUNNEL TEMPERATURE Time Delay ~~(13A-K41)~~ ~~(13A-K42)~~	≤35 minutes	Note ~~3~~ 2.d
4.c	~~2 per set of 4~~ 6 A.16 (Notes ~~11~~, ~~12~~) A.5 A.11	High Steam Line Space Temperature ~~(TS-13-(79-82)(B,C,D))~~ LA.1	~~≤212°F~~ ≤ 196 °F M.7	Note ~~3~~ 2.d
4d	1 (Notes ~~11~~, ~~12~~) A.5 A.11	High Steam Line d/p (Steam Line Break) ~~(DPIS-13-83/84)~~	≤195 inches of water	Note ~~3~~ 2.d
4.f	4 (Notes ~~8~~, ~~11~~, ~~13~~) A.5 LA.7 ~~2~~ L.4 A.11	Low Steam Supply Pressure ~~(PS-13-87(A-D))~~	≥50 psig	Note ~~3~~ 2.d
		~~Trip System Logic~~ L.4	~~--~~	~~Note 3~~ L.4
4.e	1 (Notes ~~11~~, ~~12~~) A.5 A.11	HIGH STEAM LINE d/p Time Delay ~~(13A-K7)~~ ~~(13A-K31)~~	3≤ t ≤7 seconds	Note ~~3~~ 2.d

A.1 ADD ACTION NOTE 2.a L.3 M.3

TABLE 3.2.2 ACTION NOTES

1. The main steam line low pressure need be available only in the "Run" mode. A.13

2. If the minimum number of operable instrument channels are not available for one trip system, that trip system shall be tripped. If the minimum number of operable instrument channels are not available for both trip systems, the appropriate actions listed below shall be taken: A.19

ACTION NOTE 2.b A. Initiate an orderly shutdown and have reactor in the cold shutdown condition in 24 hours. LA.4

ACTION NOTE 2.c B. Initiate an orderly load reduction and have reactor in "Hot Standby" within 8 hours.

ACTION NOTE 2.d C. Close isolation valves in system and comply with Specification 3.5. A.17 ISOLATE THE AFFECTED PENETRATION FLOW PATH WITHIN 1 HOUR M.4

4. ~~Deleted~~

5. One trip system arranged in a one-out-of-two twice logic. LA.7

6. The main steam line high flow is available only in the "Refuel," "Shutdown," and "Startup" modes. A.14

7. ~~Deleted.~~ PENETRATION FLOW PATHS MAY BE UNISOLATED UNDER ADMINISTRATIVE CONTROLS L.5

8. ~~Deleted.~~

9. ~~Deleted.~~

A.15 LA.5

10. A key lock switch is provided to permit the bypass of this trip function to enable plant startup and shutdown when the condenser vacuum is greater than 12 inches Hg absolute provided that both turbine stop and bypass valves are closed.

11. When a channel, and/or the affected primary containment isolation valve, is placed in an inoperable status solely for performance of required instrumentation surveillances, entry into associated Limiting Conditions for Operation and required ACTIONS may be delayed for up to 6 hours provided the associated Trip Function maintains isolation capability. A.5

ACTION NOTE 1 12. Whenever Primary Containment integrity is required by Specification 3.7.A.2, there shall be two operable or tripped trip systems for each Trip Function, except as provided for below: A.2 LA.6

A.18

ACTION NOTE 1.b A. With one or more automatic functions with isolation capability not maintained restore isolation capability in 1 hour or take the ACTION required by Table 3.2.2.

ACTION NOTE 1.a B. With one or more channels inoperable, place the inoperable channels (s) in the tripped condition within:

1) 12 hours for trip functions common to RPS instrumentation, and
2) 24 hours for trip functions not common to RPS instrumentation,

or, initiate the ACTION required by Table 3.2.2.

PENETRATION FLOW PATHS, ISOLATED AS A RESULT OF COMPLYING WITH THE ABOVE ACTIONS, MAY BE UNISOLATED INTERMITTENTLY UNDER ADMINISTRATIVE CONTROLS. L.5

ACTION

TABLE 3.2.2 NOTES (Cont'd)

ACTION NOTE 1 ~~13~~ Whenever the High Pressure Cooling Injection System and Reactor Core Isolation Cooling System are required to be operable in accordance with Specification 3.5, the low steam supply pressure automatic isolation trip system shall be operable except as provided below: A.2

A.18

ACTION NOTE 1.b A. With the ~~automatic~~ isolation trip function not maintained, restore isolation capability in 1 hour or take the ACTION required by Table 3.2.2.

ACTION NOTE 1.a B. With one or more required channels inoperable, place the inoperable channel(s) in the tripped condition within 24 hours or take the ACTION required by Table 3.2.2. M.5

PENETRATION FLOW PATHS, ISOLATED AS A RESULT OF COMPLYING WITH THE ABOVE ACTIONS, MAY BE UNISOLATED INTERMITTENTLY UNDER ADMINISTRATIVE CONTROLS.

L.5

A.1

PRIMARY CONTAINMENT ISOLATION A.7

TABLE 4.2.~~1~~ 2
(Cont'd)

~~MINIMUM~~ TES~~TS~~ AND ~~CALIBRATION~~ FREQUENCIES

~~EMERGENCY CORE COOLING ACTUATION~~ INSTRUMENTATION

~~Low Pressure Coolant Injection System~~

Trip Function	Functional Test (8) A.21	Calibration (8) A.21	Instrument Check
Low Reactor Pressure (PT-2-3-56C/D(M))	Every Three Months	Once/Operating Cycle	--
High Drywell Pressure (PT-10-101A-D(M))	Every Three Months	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	Every Three Months	Once/Operating Cycle	Once Each Day
Reactor Vessel Shroud Level	Every Three Months	Once/Operating Cycle	--
5.a ~~Low~~ High A.6 Reactor Pressure ~~(PS-2-128A/B)~~ LA.1	Every Three Months	Every Three Months	--
RHR Pump Discharge Pressure	Every Three Months	Every Three Months	--
High Drywell Pressure (PT-10-101A-D(S1))	Every Three Months	Once/Operating Cycle	--
Low Reactor Pressure (PT-2-3-56A/B)(M) & 52C/D(M))	Every Three Months	Once/Operating Cycle	--
Auxiliary Power Monitor	Every Three Months	None	Once Each Day
Pump Bus Power Monitor	Every Three Months	None	Once Each Day
SR 4.2.B.2 Trip System Logic	Once/Operating Cycle	~~Once/Operating Cycle (Note 3)~~ A.20	--

A.9

A.9

A.3

A.1

A.10

TABLE 4.2.2

~~MINIMUM~~ TES~~TS~~ AND ~~CALIBRATION~~ FREQUENCIES

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

A.21

	Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
1.a	Low-Low Reactor Vessel Water Level	Every Three Months	Once/Operating Cycle	Once Each Day
1.b	High MAIN Steam Line Area Temperature	Every Three Months	Each Refueling Outage EVERY 3 MONTHS (1),	--
1.c, 1.e	High MAIN Steam Line Flow " " " " " — NOT IN RUN	Every Three Months	Once/Operating Cycle	Once Each Day
1.d	Low Main Steam Line Pressure	Every Three Months	Every Three Months	--
2.a	Low Reactor Vessel Water Level	Every Three Months	Once/Operating Cycle	--
2.b	High Drywell Pressure	Every Three Months	Once/Operating Cycle	Once Each Day
1.f	Condenser Low Vacuum	Every Three Months	Every Three Months	--
SR 4.2.B.2	Trip System Logic	Once/Operating Cycle ~~(Note 2)~~	~~Once/Operating Cycle (Note 3)~~	--

M.6

A.3

LA.8

A.20

(1) TRIP UNIT CALIBRATION ONLY   M.6

A.1

TABLE 4.2.2
(Cont'd)

~~MINIMUM~~ TEST~~S~~ AND ~~CALIBRATION~~ FREQUENCIES

HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

	Trip Function	Functional Test ~~(8)~~ A.21	Calibration ~~(8)~~ A.21	Instrument Check
3.a	High Steam Line Space Temperature	Every Three Months	Each refueling outage	--
3.b	High Steam Line D/P (Steam Line Break)	Every Three Months	Every three months	--
3.c	Low HPCI Steam Supply Pressure	Every Three Months	Every three months	--
3.d	Main Steam Line Tunnel Temperature	Every Three Months	Each refueling outage	--
SR4.2.B.2	Trip System Logic	Once/operating cycle	Once/operating cycle ~~(Note 3)~~	-- A.3
3.e	High Main Line Tunnel Temperature Time Delay		A.22	

A.1

TABLE 4.2.2
(Cont'd)

~~MINIMUM~~ TESTS AND ~~CALIBRATION~~ FREQUENCIES

REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

	Trip Function	Functional Test ~~(8)~~ A.21	Calibration ~~(8)~~ A.21	Instrument Check
4.a	High Main Steam Line Tunnel Temperature	Every Three Months	Each refueling outage	--
4.c	High Steam Line Space Temperature	Every Three Months	Each refueling outage	--
4.d 4.e	High Steam Line d/p including time delay relays (Steam Line Break)	Every Three Months	Every three months	--
4.f	Low RCIC Steam Supply Pressure	Every Three Months	Every three months	--
SR4.2.B.2	Trip System Logic	Once/operating cycle	Once/operating cycle ~~(Note 3)~~	-- A.3
4.b	High Main Steam Line Tunnel Temperature Time Delay		A.22	

A.1

TABLE 4.2 NOTES

LA.8

1. Not used.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart.

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system. A.22

4. This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. A.9

5. Deleted.

6. Deleted.

7. Deleted.

A.21

8. Functional tests and calibrations are not required when systems are not required to be operable.

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

12. Trip system logic testing is not applicable to this function. If the required surveillance frequency (every Refueling Outage) is not met, functional testing of the Reactor Mode Switch-Shutdown Position function shall be initiated within 1 hour after the reactor mode switch is placed in Shutdown for the purpose of commencing a scheduled Refueling Outage.

13. Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).

A.9

Safety Assessment

Discussion of Changes

3.2.B and 4.2.B

Primary Containment Isolation

ADMINISTRATIVE

A.1 In the revision of the Vermont Yankee Nuclear Power Station (VYNPS) current Technical Specifications (CTS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the VYNPS Technical Specifications (TS) more consistent with human factor principles used in the Boiling Water Reactor Improved Standard Technical Specifications (ISTS), NUREG-1433, Rev. 2. These format and presentation changes are being made to improve usability and clarity. The changes are considered administrative.

A.2 CTS 3.2.B specifies an Applicability for primary containment isolation instrumentation of "When primary containment is required, in accordance with Specification 3.7." CTS Table 3.2.2 Note 12 specifies an Applicability of "Whenever Primary Containment integrity is required by Specification 3.7.A.2." CTS Table 3.2.2 Note 13 specifies an Applicability of "Whenever the High Pressure Cooling Injection System and Reactor Core Isolation Cooling System are required to be operable in accordance with Specification 3.5" for the High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System Low Steam Supply Pressure Trip Functions. Specification 3.5 provides requirements for the HPCI and RCIC Systems. Specification 3.7 includes the requirements for the primary containment. This change provides an explicit Applicability, in proposed Table 3.2.2 for each primary containment isolation instrumentation trip function. The specified Applicabilities, in proposed Table 3.2.2, are consistent with the Modes and conditions when primary containment integrity is required to be operable by Specification 3.7, and, for the HPCI System and RCIC System Low Steam Supply Pressure Trip Functions, when HPCI and RCIC are required to be operable by Specification 3.5; except as provided and justified in change L.1 below. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. The change, providing explicit Mode or conditions of Applicability for each trip function, is consistent with the ISTS.

A.3 CTS 4.2.B specifies that instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.2. In proposed Surveillance Requirement (SR) 4.2.B.1, the reference to "and logic system," is deleted since associated logic systems are considered part of the primary containment isolation instrumentation Trip Functions in both proposed and CTS Tables 3.2.2 and 4.2.2. It is not necessary to explicitly identify logic systems in CTS 4.2.B, since proposed SR 4.2.B.2 (CTS Table 4.2.2 requirements to perform Functional Tests of Trip System Logic) continues to require performance of surveillance testing of Trip System Logic (i.e., performance of Logic System Functional Tests for each primary containment isolation instrumentation Trip Function). Therefore, this change is considered administrative.

ADMINISTRATIVE (continued)

A.4 CTS 4.2.B includes reference to CTS Table 4.2.2 for functional test and calibration requirements for primary containment isolation instrumentation. CTS 4.2.B is revised, in proposed SR 4.2.B.1, to also include reference to check requirements consistent with CTS Table 4.2.2. This change is a presentation preference and does not alter the current requirements to periodically perform checks of certain primary containment isolation instrument trip functions. Therefore, this change is considered administrative in nature.

A.5 CTS Table 3.2.1, Note 9 and CTS Table 3.2.2 Note 11, provide allowances to delay entry into actions for 6 hours for the situation of a channel inoperable solely for performance of surveillances. These allowances are moved to proposed SR 4.2.B.1 and the allowances of these two notes are combined. This change does not involve a technical change, but is only a difference of presentation preference. Therefore, this change is considered administrative.

A.6 CTS Table 3.2.1 includes a Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function (PS-2-128A & B). The purpose of this trip function is to provide isolation of the shutdown cooling portion of the Residual Heat Removal (RHR) System to protect that system from overpressurization due to high reactor pressure. This isolation provides for equipment protection to prevent an intersystem LOCA scenario. As a result, the name of this trip function is revised to "Residual Heat Removal Shutdown Cooling Isolation - High Reactor Pressure" in proposed Table 4.2.2 to more accurately reflect its function. The design and operation of the actual instrumentation is unchanged. Therefore, this change is considered administrative.

A.7 CTS Tables 3.2.1 and 4.2.1 include requirements for the Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function (PS-2-128A & B) and associated Trip System Logic. The purpose of this Trip Function and associated Trip System Logic is to provide isolation of the shutdown cooling portion of the Residual Heat Removal (RHR) System to protect that system from overpressurization due to high reactor pressure. This isolation provides for equipment protection to prevent an intersystem LOCA scenario. These requirements are to be moved to the primary containment isolation instrumentation TS. Given the isolation function of this Trip Function and associated Trip System Logic, they are more appropriately located in proposed Tables 3.2.2 and 4.2.2. As a corresponding change, the reference to initiation capability in CTS Table 3.2.1 Note 10 (proposed Table 3.2.2 Action Note 1) is editorially changed to isolation capability. This change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. The change, including the requirements this Trip Function and associated Trip System Logic in the primary containment isolation instrumentation TS, is consistent with the ISTS.

ADMINISTRATIVE (continued)

A.8 Notes 10 and 5 to CTS Table 3.2.1 provide actions when the minimum number of channels per trip system requirement is not met. These requirements are divided and identified in two separate columns in proposed Table 3.2.2 titled, "ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE" and "ACTIONS REFERENCED FROM ACTION NOTE 1." This change is a presentation preference and does not alter the current action requirements when the required channels are inoperable. Therefore, this change is considered administrative in nature.

A.9 CTS 3.2.1 and 4.2.1 and associated Notes provide requirements related to Recirculation Pump Trip instrumentation. The requirements applicable to the Recirculation Pump Trip instrumentation are physically moved and changes addressed in proposed Specifications 3.2.I and 4.2.I. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS 3.2.1 and 4.2.1 and associated Notes also provide requirements for ECCS Instrumentation. The ECCS instrumentation requirements are included in proposed Tables 3.2.1 and 4.2.1. Changes to the ECCS instrumentation requirements are addressed in the safety assessment of changes for TS 3.2.A/4.2.A, ECCS Instrumentation. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Notes 4, 12, and 13 provides requirements that apply to control rod block instrumentation. The control rod block instrumentation is located in proposed Specifications 3.2.E and 4.2.E. Therefore, the requirements of CTS Table 4.2 Note 4 are physically moved and changes addressed in proposed Specifications 3.2.E and 4.2.E. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Note 9 provides requirements that apply to post-accident monitoring instrumentation. The post-accident monitoring instrumentation is located in proposed Specifications 3.2.G and 4.2.G. Therefore, the requirements of CTS Table 4.2 Note 9 are physically moved and changes addressed in proposed Specifications 3.2.G and 4.2.G. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.10 The Primary Containment Isolation Instrumentation portion of CTS Tables 3.2.2 and 4.2.2 is divided into two sections, Main Steam Line Isolation (Trip Function 1), and Primary Containment Isolation (Trip Function 2) in proposed Tables 3.2.2 and 4.2.2. The appropriate individual trip functions are placed with the proper isolation. Since the current requirements are maintained (except as addressed in the changes below), the change is considered to be administrative in nature. This change is consistent with the ISTS.

A.11 Note 12 to CTS Table 3.2.2 provides actions when the minimum number of channels per trip system requirement is not met. These requirements are identified in a separate column in proposed Table 3.2.2 titled, "ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE." This change is a presentation preference and does not alter the current action requirements when required primary containment isolation instrumentation channels are inoperable. Therefore, this change is considered administrative in nature.

ADMINISTRATIVE (continued)

A.12 The name of the CTS Table 3.2.2 High Main Steam Line Flow Trip Function (DPT-2-116A, 117B, 118C, 119D (S1)) is revised to reflect the condition when this trip function is available. This trip function is available only in the Refuel, Shutdown, and Startup Modes (i.e., not available in Run). Therefore, the name of proposed Tables 3.2.2 and 4.2.2 Trip Function 1.e is "High Main Steam Line Flow – Not in RUN." The design and operation of the actual instrumentation is unchanged. Therefore, this change is considered administrative.

A.13 CTS 3.2.B requires that the instrumentation in CTS Table 3.2.2 be operable when primary containment integrity is required in accordance with Specification 3.7. CTS 3.7 requires primary containment integrity when reactor water temperature is above 212°F and fuel is in the reactor vessel. The Primary Containment Isolation Instrumentation Low Main Steam Line Pressure Trip Function requirements of CTS Table 3.2.2 are modified by CTS Table 3.2.2 Note 1. CTS Table 3.2.2 Note 1 states that the main steam line low pressure need be available only in the Run Mode. The intent of this note is to waive the operability requirements of the Primary Containment Isolation Instrumentation Low Main Steam Line Pressure Trip Function when the reactor is not in the Run Mode. The "APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS" column in proposed Table 3.2.2 requires the Low Main Steam Line Pressure Trip Function (proposed Table 3.2.2, Trip Function 1.d) to be operable in the Run Mode which is equivalent to CTS requirements. As such, this change is considered administrative in nature. This change is consistent with the ISTS.

A.14 CTS 3.2.B requires that the instrumentation in CTS Table 3.2.2 be operable when primary containment integrity is required in accordance with Specification 3.7. CTS 3.7 requires primary containment integrity when reactor water temperature is above 212°F and fuel is in the reactor vessel. The Primary Containment Isolation Instrumentation High Main Steam Line Flow Trip Function (DPT-2-116A, 117B, 118C, 119D (S1)) requirements of CTS Table 3.2.2 are modified by CTS Table 3.2.2 Note 6. CTS Table 3.2.2 Note 6 states that the main steam line high flow is available only in the Refuel, Shutdown, and Startup Modes. The intent of this note is to waive the operability requirements of the Primary Containment Isolation Instrumentation High Main Steam Line Flow Trip Function (DPT-2-116A, 117B, 118C, 119D (S1)) when the reactor is in the Run Mode. The "APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS" column in proposed Table 3.2.2 requires the High Main Steam Line Flow Trip Function (proposed Table 3.2.2, Trip Function 1.e) to be operable in the Startup/Hot Standby, Hot Shutdown, and Refuel (with reactor coolant water temperature > 212°F) Modes which is equivalent to CTS requirements. As such, this change is considered administrative in nature.

ADMINISTRATIVE (continued)

A.15 CTS 3.2.B requires that the instrumentation in CTS Table 3.2.2 be operable when primary containment integrity is required in accordance with Specification 3.7. CTS 3.7 requires primary containment integrity when reactor water temperature is above 212°F and fuel is in the reactor vessel. The Primary Containment Isolation Instrumentation Condenser Low Vacuum Trip Function requirements of CTS Table 3.2.2 are modified by CTS Table 3.2.2 Note 10. CTS Table 3.2.2 Note 10 states "A key lock switch is provided to permit the bypass of this trip function to enable plant startup and shutdown when condenser vacuum is greater than 12 inches Hg absolute provided that both turbine stop and bypass valves are closed." The intent of this note is to waive the operability requirements of the Primary Containment Isolation Instrumentation Condenser Low Vacuum Trip Function when the reactor is not in the Run Mode and all turbine stop and bypass valves are closed. The "APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS" column in proposed Table 3.2.2 requires the Condenser Low Vacuum Trip Function (proposed Table 3.2.2, Trip Function 1.f) to be operable in the Run, Startup/Hot Standby (with any turbine stop valve or turbine bypass valve not closed), Hot Shutdown (with any turbine stop valve or turbine bypass valve not closed), and Refuel (with reactor coolant water temperature > 212°F and with any turbine stop valve or turbine bypass valve not closed) Modes which is equivalent to CTS requirements. As such, this change is considered administrative in nature.

A.16 All HPCI System Isolation High Steam Line Space channels are required to be operable to assure isolation with the worst single failure. CTS Table requires a minimum of 2 per set of 4 channels per trip system of the HPCI System Isolation High Steam Line Space Trip Function (proposed Table 3.2.2, Trip Function 3.a) to be operable. There are three locations (i.e., 3 sets), each monitored by one set of 4 channels. As a result, there are a total of 12 channels for this trip function, with 6 channels per trip system. Therefore, the minimum number of channels per trip system required to be operable for this trip function is specified as "6" in proposed Table 3.2.2. Since this change involves no design change but is only a difference of nomenclature and presentation preference, this change is considered administrative.

A.17 The CTS Table 3.2.2 Note 3 action requirement to "comply with Specification 3.5" is an unnecessary reminder that other Technical Specifications may be affected when isolation valves in the HPCI System or RCIC System are closed. This is essentially a "cross reference" between Technical Specifications that has been determined to be adequately provided through training. Therefore, the deletion is considered to be administrative. This change is consistent with the ISTS.

A.18 CTS Table 3.2.2 Notes 12 and 13 refer to automatic isolation trip functions. However, CTS Table 3.2.2 (proposed Table 3.2.2) includes only automatic isolation trip functions. Manual isolation instrumentation trip functions are not included in the VYNPS CTS. Therefore, it is unnecessary to use the word "automatic" when referring to CTS Table 3.2.2 isolation trip functions in proposed Table 3.2.2 Action Note 1. This change is a presentation preference

ADMINISTRATIVE

A.18 (continued) and does not alter the current action requirements when required primary containment isolation instrumentation channels are inoperable. Therefore, this change is considered administrative in nature.

A.19 CTS Table 3.2.2 Note 2 states, in the first paragraph, "If the minimum number of operable instrument channels are not available for one trip system, that trip system shall be tripped. If the minimum number of operable instrument channels are not available for both trip systems, the appropriate actions listed below shall be taken..." However, due to the presentation of the Notes in CTS Table 3.2.2, Note 2 actions are only taken after actions associated with CTS Table 3.2.2 Note 12 or 13, as applicable, are taken. Since the CTS Table 3.2.2 Note 12 and 13 actions (proposed Table 3.2.2 ACTION Note 1) already provide the appropriate NRC approved actions for each of the conditions addressed in the first paragraph of CTS Table 3.2.2 Note 2, the first paragraph of CTS Table 3.2.2 Note 2 is unnecessary and is deleted. Since actions when required primary containment isolation instrumentation channels are inoperable will continue to be taken in the same manner and in the same time period, the deletion is considered administrative in nature.

A.20 For the Trip System Logic associated with the Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function (PS-2-128A & B), CTS Table 4.2.1 includes a requirement to perform a calibration of Trip System Logic once per Operating Cycle. For the Trip System Logic associated with the Primary Containment Isolation Instrumentation, CTS Table 4.2.2 includes a requirement to perform a calibration of Trip System Logic once per Operating Cycle. These requirements are modified by Table 4.2 Note 3. Note 3 states, "Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system." The Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function instrumentation of CTS 4.2.1 and the Primary Containment Isolation Instrumentation Trip Functions of CTS 4.2.2 do not include any time delay relays or timers necessary for proper functioning of the trip systems. Therefore, this Note is deleted and, in proposed Table 4.2.2, the Residual Heat Removal Shutdown Cooling Isolation instrumentation (proposed Trip Function 5.a), Main Steam Line Isolation instrumentation (proposed Trip Functions 1.a through 1.f) and Primary Containment Isolation instrumentation (proposed Trip Functions 2.a and 2.b) do not include calibration requirements for time delay relays or timers. As a result, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.21 CTS Table 4.2 Note 8 states that functional tests and calibrations are not required when systems are not required to operable. The requirements of this Note are duplicated in the CTS definition 1.0.Z, "Surveillance Interval," which states that these tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but that these tests shall be performed on the instrument, component, or system prior to being required to be operable. Therefore, CTS Table 4.2 Note 8 is unnecessary and its deletion is considered to be administrative. The change is consistent with the ISTS.

ADMINISTRATIVE (continued)

A.22 CTS Table 4.2.2 includes a requirement to perform a calibration of Trip System Logic once per Operating Cycle. This requirement is modified by Table 4.2 Note 3. Note 3 states, "Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system." In proposed Table 4.2.2, this requirement is reflected with explicit requirements to perform calibrations of the required HPCI System Isolation and RCIC System Isolation instrumentation time delay relays and timers (i.e., proposed Table 4.2.2 Trip Function 3.e., HPCI System Isolation – High Main Steam Tunnel Temperature Time Delay, and Trip Function 4.b, RCIC System Isolation – High Main Steam Tunnel Temperature Time Delay) once per Operating Cycle. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Table 3.2.1 includes a Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function (PS-2-128A & B). The purpose of this trip function is to provide isolation of the shutdown cooling portion of the Residual Heat Removal (RHR) System to protect that system from overpressurization due to high reactor pressure. This isolation provides for equipment protection to prevent an intersystem LOCA scenario. The associated action for this Trip Function, in the event the instrumentation is inoperable and not restored within the allowed time period, in CTS Table 3.2.1 Note 10 requires that the associated systems to be declared inoperable. Continued operation, in this condition, is then allowed for a limited duration in accordance the associated system TS. However, the TS actions for Low Pressure Coolant Injection System inoperabilities do not require that overpressure protection of the RHR System be provided. Therefore, in the same condition, proposed Table 3.2.2 Action Note 2.d will require the associated penetration to be isolated within one hour, thus restoring overpressure protection for the RHR System. This change represents an additional restriction on plant operation necessary to provide overpressure protection to prevent an intersystem LOCA scenario. The change is consistent with the ISTS.

M.2 CTS Table 3.2.2 provides requirements for the Primary Containment Isolation Instrumentation - High Main Steam Line Area Temperature Trip Function. CTS Table 3.2.2 specifies that, for this trip function, the minimum number of operable instrumentation channels per trip system is 2 of 4 in each of 2 channels. There are 2 trip systems for this trip function, each with 2 channels. As such, the CTS require that only 2 temperature sensor inputs (out of 4) per channel be operable per trip system. Therefore, the total number of temperature sensors required to be operable by CTS Table 3.2.2 is 8 (i.e., 2/channel x 2 channels/trip system x 2 trip systems). The temperature sensor arrangement in the channels is such that each of the 4 sensors in a channel monitors temperature of a different area of the main steam lines. This arrangement, when combined with the CTS allowance for any two sensors in one or more logic channels to be inoperable without affecting compliance with the Limiting Conditions for Operation, results in the possibility for a loss of isolation capability for a leak in a specific main steam line area. Therefore, the

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (continued) required number of number of operable channels per trip system for the High Main Steam Line Area Temperature Trip Function (proposed Table 3.2.2 Trip Function 1.b) is revised to 8 per trip system (for a total number of temperature sensors required to be operable of 16). This change represents an additional restriction on plant operation to ensure adequate temperature monitoring of the main steam line areas is provided.

M.3 CTS 3.2.B requires that the Primary Containment Isolation Instrumentation High Main Steam Line Area Temperature Trip Function (proposed Table 3.2.2 Trip Function 1.b) and the High Main Steam Line Flow (DPT-2-(116-119) (A-D) (M)) Trip Function (proposed Table 3.2.2 Trip Function 1.c) be operable in Run, Startup/Hot Standby, Hot Shutdown, and Refuel (with reactor coolant water temperature > 212°F) Modes and that the Primary Containment Isolation Instrumentation High Main Steam Line Flow (DPT-2-116A, 117B, 118C, 119D (S1)) Trip Function (proposed Table 3.2.2 Trip Function 1.e) be operable in Startup/Hot Standby, Hot Shutdown, and Refuel (with reactor coolant water temperature > 212°F) Modes. In the event minimum conditions for operation are not satisfied for these trip functions, CTS Table 3.2.2 requires compliance with CTS Table 3.2.2 Note 2.B. CTS Table 3.2.2 Note 2.B requires an orderly load reduction to be initiated and to the have the reactor in Hot Standby within 8 hours. However, this action does not result in satisfying the isolation function of the inoperable instrumentation. Nor does this action result in placing the reactor in a Mode in which the instrumentation is not required to be operable to provided isolation. As a result, CTS would allow continued operation with the isolation function of this instrumentation not maintained. Therefore, proposed Table 3.2.2 Action Note 2.a will require, in the same condition, the associated main steam line to be isolated within 12 hours (which satisfies the isolation function of the inoperable instrumentation) or that the reactor be placed in Hot Shutdown within 12 hours and in Cold Shutdown within the next 12 hours (which places the reactor in a Mode in which the instrumentation is not required to be operable). The time period specified for exiting the Modes of Applicability are consistent with times to reach the same Modes provided in other VYNPS TS. This change represents an additional restriction on plant operation necessary to ensure that either the isolation function of the inoperable instrumentation is satisfied or that the reactor is placed in a Mode in which the instrumentation is not required to function to provide isolation. The change is consistent with the ISTS.

M.4 CTS Table 3.2.2 Note 3 effectively specifies actions to isolate the affected penetration flow path (i.e., close isolation valves in the system) but does not state the time period in which this action is to be completed. Proposed Table 3.2.2 Action Note 2.d provides a one hour time period in which to complete the action of isolating the affected penetration flow. Placing a limitation on the time allowed to complete the associated actions represents an additional restriction on plant operation since the time period allowed to complete the isolation will be controlled through TS. This change is consistent with the ISTS.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.5 CTS Table 3.2.2 Note 13.B provides actions for inoperable time delay channels for HPCI and RCIC Isolation instrumentation. These actions allow the inoperable channel to be tripped, rather than requiring the channel to be restored to operable status. The subject channels are associated with Time Delay Trip Functions. As such, placing these Time Delay Trip Functions channels in trip (as allowed by Note 13.B) does not perform the intended function (providing a time delay for isolation after certain conditions are satisfied) and could adversely impact the capability of the HPCI or RCIC Systems to perform their intended functions. Therefore, the actions for inoperable channels of these Time Delay Trip Functions (proposed Table 3.2.2 Trip Functions 3.e, HPCI System Isolation – High Main Steam Line Tunnel Temperature Time Delay, 4.b, RCIC System Isolation – High Main Steam Line Tunnel Temperature Time Delay, and RCIC System Isolation – High Steam Line d/p Time Delay) are revised, in proposed Table 3.2.2 Action Note 1.a.2), to require that they be restored to operable status rather than being placed in trip. This change represents an additional restriction on plant operation and is consistent with the ISTS (i.e., inoperable time delays are required to be restored to operable status rather than placed in trip).

M.6 CTS Table 4.2.2 does not include explicit requirements to calibrate trip units. Proposed Table 4.2.2 requires calibration of the trip units of the following Trip Functions every 3 months: Main Steam Line Isolation – Low-Low Reactor Vessel Water Level (proposed Table 4.2.2 Trip Function 1.a); Main Steam Line Isolation – High Steam Line Flow (proposed Table 4.2.2 Trip Function 1.c); Main Steam Line Isolation – High Steam Line Flow-Not in Run (proposed Table 4.2.2 Trip Function 1.e); and Primary Containment Isolation – High Drywell Pressure (proposed Table 4.2.2 Trip Function 2.b). The trip units of these Trip Functions are currently required by CTS Table 4.2.2 to be calibrated with the rest of the associated instrument loops once per operating cycle. Therefore, this change is more restrictive. This change is necessary to ensure consistency with assumptions regarding trip unit calibration frequency used in the associated setpoint calculations. This change is consistent with the ISTS.

M.7 CTS Table 3.2.2 specifies for the Primary Containment Isolation High Main Steam Line Area Temperature, the HPCI System Isolation High Steam Line Space Temperature, the HPCI System Isolation Main Steam Line Tunnel Temperature, the RCIC System Isolation Main Steam Line Tunnel Temperature, and the RCIC System High Steam Line Space Temperature Trip Functions that the Trip Settings be $\leq$ 212°F. The function of these instruments is to provide isolation in the event of breaks in the associated steam lines. The CTS Trip Settings have been determined to be insufficient to ensure isolation occurs as assumed in the high energy line break and Equipment Qualification (EQ) Program analyses. Therefore, in proposed Table 3.2.2, the Trip Setting for the Primary Containment Isolation High Main Steam Line Temperature Area Temperature Trip Function (Trip Function 1.b) has been decreased to $\leq$ 196°F for channels monitoring outside the steam tunnel and $\leq$ 200°F for channels monitoring inside the steam tunnel; the Trip Settings for the HPCI System Isolation and RCIC System Isolation High Steam Line Space Temperature Trip Functions (Trip Functions 3.a and 4.c, respectively) have been

TECHNICAL CHANGES - MORE RESTRICTIVE

M.7 (continued) decreased to ≤ 196°F; and the Trip Settings for the HPCI System Isolation and RCIC System Isolation Main Steam Line Tunnel Temperature Trip Functions (Trip Functions 3.d and 4.a, respectively) have been decreased to ≤ 200°F. These revised Trip Settings are consistent the assumptions of the high energy line break and EQ Program analyses and correspond to the Analytical Limit used in the associated setpoint calculations. To account for instrument uncertainties, the instrument setpoints and as-found tolerances (i.e., instrument operability limits) were developed using the Vermont Yankee Instrument Uncertainty and Setpoints Design Guide. The instrument setpoints and as-found tolerances are located in plant procedures. This change represents an additional restriction on plant operation necessary to ensure that isolation of the associated steam lines occurs as assumed in analyses.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Tables 3.2.1, 3.2.2, 4.2.1 and 4.2.2 details relating to system design and operation (i.e., the specific instrument tag numbers) are unnecessary in the TS and are proposed to be relocated to the Technical Requirements Manual (TRM). Proposed Specification 3.2.B and Table 3.2.2 require the Primary Containment Isolation Instrumentation Trip Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.2 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required Primary Containment Isolation Instrumentation Trip Functions are maintained operable. As such, the relocated details are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TRM are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.2 CTS Table 3.2.1 includes requirements for the Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function (PS-2-128A & B). The purpose of this trip function is to provide isolation of the shutdown cooling portion of the Residual Heat Removal (RHR) System to protect that system from overpressurization due to high reactor pressure. This isolation provides for equipment protection to prevent an intersystem LOCA scenario. CTS Table 3.2.1 includes Trip Settings of $100 \leq p \leq 150$ psig. The upper Trip Setting ensures the RHR System is isolated from the Reactor Coolant System prior to being overpressurized due to high reactor pressure. The lower Trip Setting is an operational detail that is not directly related to the operability of the associated instrumentation. This detail is to be relocated to plant procedures. The upper Trip Setting is the required limitation for the parameter and this value is retained in the VYNPS TS. As such, the lower Trip Setting for this trip function is not required to be in TS to provide adequate protection of the public health and safety. Changes to the relocated lower Trip Setting in the plant procedures will be controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.3 The Trip Setting associated with reactor vessel water level trip function (proposed Table 3.2.2 Trip Function 1.a) is currently referenced to "above the top of enriched fuel." This detail is to be relocated to the Bases. This reference is not necessary to be included in the VYNPS TS to ensure the operability of the associated primary containment isolation instrumentation. Operability requirements are adequately addressed in proposed Specification 3.2.B, Table 3.2.2 and the specified Trip Setting. As such, this relocated reference is not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TS Bases are controlled by 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.4 Details of the methods for performing CTS Table 3.2.2 Notes 2.A and 2.B (proposed Table 3.2.2 Action Notes 2.b and 2.c), associated with placing the reactor in Cold Shutdown (i.e., initiating an orderly shutdown) or Hot Standby (i.e., initiating an orderly load reduction), are to be relocated to plant procedures. These details are not necessary to ensure the actions of placing the reactor in Cold Shutdown mode or Hot Standby and exiting the applicable Mode of the associated primary containment isolation instrumentation is accomplished. The requirements of proposed Table 3.2.2 and Table 3.2.2 Action Notes 2.b and 2.c are adequate to ensure this action is accomplished. As such, these relocated details are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the plant procedures are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.5 For the Primary Containment Isolation Instrumentation – Condenser Low Vacuum Trip Function, CTS Table 3.2.2 Note 5 states that "A key lock switch is provided to permit the bypass of this trip function to enable plant startup and shutdown when condenser vacuum is greater than 12 inches Hg absolute provided that both turbine stop and bypass valves are closed." The system design details in CTS Table 3.2.2 Note 5 are to be relocated to the Bases and the reference to this information is deleted from the VYNPS TS. These design details are not necessary to be included in the TS to ensure the operability of the Condenser Low Vacuum Trip Function instrumentation since operability requirements are adequately addressed in proposed Specification 3.2.B and Table 3.2.2. Therefore, these relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.6 CTS Table 3.2.2 Note 12 contains design and operational details of the primary containment isolation instrumentation (i.e., there shall be two operable or tripped trip systems for each Trip Function). These details are not necessary to ensure the operability of associated primary containment isolation instrumentation. Therefore, the information in this note is to be relocated to Specification 3.2.B Bases and reference to this information is deleted from VYNPS TS. The requirements of Specification 3.2.B and the associated Surveillance Requirements for the associated primary containment isolation instruments are adequate to ensure the instruments are maintained operable. As such, these relocated requirements are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TS Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.7 CTS Table 3.2.2 Note 5 contains design details of the HPCI System Isolation – Low Steam Supply Pressure Trip Function and RCIC System - Low Steam Supply Pressure Trip Function instrumentation (i.e., one trip system arranged in a one-out-of-two twice logic). These details are not necessary to ensure the operability of associated isolation instrumentation. Therefore, the information in these notes is to be relocated to Specification 3.2.B Bases and reference to this information is deleted from VYNPS TS. The requirements of Specification 3.2.B and the associated Surveillance Requirements for these isolation instruments are adequate to ensure the instruments are maintained operable. As such, these relocated requirements are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TS Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.8 CTS Table 4.2.2 and associated Note 2 describe details of the performance of the Functional Test of the Trip System Logic associated with the Main Steam Line Isolation Trip Functions. These details are to be relocated to Bases. These details are not necessary to ensure the operability of the associated Trip System Logic instrumentation. The VYNPS TS definition of Logic System Functional Tests, the requirements of proposed Specification 3.2.B, and the associated Surveillance Requirements (including the requirements to periodically perform Logic System Functional Tests) are adequate to ensure the associated Trip System Logic is maintained operable. As such, these relocated details are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

"Specific"

L.1 CTS Tables 3.2.1 and 4.2.1 include requirements for the Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function (PS-2-128A & B) and associated Trip System Logic. The purpose of this Trip Function and associated Trip System Logic is to provide isolation of the shutdown cooling portion of the Residual Heat Removal (RHR) System to protect that system from overpressurization due to high reactor pressure. This

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (continued) isolation provides for equipment protection to prevent an intersystem LOCA scenario. CTS 3.2.A specifies an Applicability for this Trip Function and associated Trip System Logic of "When the system(s) it initiates or controls is required in accordance with Specification 3.5." Specification 3.5 requires the Low Pressure Coolant Injection (LPCI) System to be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from cold shutdown. Specification 3.5 also allows ECCS (including LPCI) to be inoperable in Cold Shutdown and Refueling provided no operations with the potential for draining the reactor vessel are in progress. The "APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS" column in proposed Table 3.2.2 requires the Residual Heat Removal Shutdown Cooling Isolation Trip Function (i.e., Trip Function 5.a) to be operable in the Run, Startup/Hot Standby, Hot Shutdown, and Refuel (with reactor coolant water temperature > 212°F) Modes since these are the only Modes in which the reactor can be pressurized such that isolation of the RHR System is required to prevent overpressurization. Therefore, the requirements for the Residual Heat Removal Shutdown Cooling Isolation Trip Function in Cold Shutdown or Refueling (with reactor coolant water temperature ≤ 212°F) are not necessary to protect the RHR System from overpressurization due to high reactor pressure and are deleted. This change is consistent with the ISTS.

L.2 CTS Table 3.2.1 Note 10.A requires that associated systems be declared inoperable within 1 hour of discovery of loss of initiation capability for feature(s) in one division when Low Pressure Coolant Injection System - Low Reactor Pressure Trip Function channels are inoperable. This Note was intended to provide requirements to ensure that a complete loss of function (in this case, loss of capability to isolate the penetration with at least one isolation valve) does not exist (for more than 1 hour) due to more than one instrument channel of an individual Trip Function being inoperable. However, the subject action requirements were written to require the implementation of the more restrictive allowed outage times associated with a loss of function (i.e., 1 hour) even for conditions for which safety function was maintained. As an example, for a Trip Function with two trip systems each providing isolation signals to two isolation valves in a penetration flow path, if the isolation capability associated with one of the two isolation valves is inoperable, the instrumentation inoperability can be such that the automatic isolation of the penetration by the second isolation valve can still be accomplished by the Trip Function. Therefore, a complete loss of function has not occurred and it is not appropriate to apply the more restrictive loss of function allowed outage time of 1 hour for this condition. Consistent with the intent of these "loss of function" requirements, proposed Table 3.2.2 Action Note 1.b is revised to require, with isolation capability not maintained, that isolation capability be restored within one hour. This change is acceptable, since if sufficient instrument channels are operable or in trip such that a loss of isolation capability has not occurred, the allowed outage times of proposed Table 3.2.2 Action Note 1.a will limit operation in this condition to within the bounds of the applicable analysis, i.e., NEDC-31677-P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation," July 1990. This change is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.3 CTS 3.2.B provides requirements for the Primary Containment Isolation Instrumentation Low-Low Reactor Vessel Water Level Trip Function (proposed Table 3.2.2 Trip Function 1.a) and the Primary Containment Isolation Instrumentation Condenser Low Vacuum Trip Function (proposed Table 3.2.2 Trip Function 1.f). In the event minimum conditions for operation are not satisfied for these trip functions, CTS Table 3.2.2 requires compliance with CTS Table 3.2.2 Note 2.A. CTS Table 3.2.2 Note 2.A requires an orderly load reduction to be initiated and to the have the reactor in Cold Shutdown within 24 hours. The function of this instrumentation is to provide main steam line isolation. Therefore, proposed Table 3.2.2 Action Note 2.a will require, in the same condition, the associated main steam line to be isolated within 12 hours (which satisfies the isolation function of the inoperable instrumentation) or that the reactor be placed in Hot Shutdown within 12 hours and in Cold Shutdown within the next 12 hours (which places the reactor in a Mode in which the instrumentation is not required to be operable). The total time period specified for exiting the Modes of Applicability is consistent with times to reach the same Modes provided in CTS Table 3.2.2 Note 2.A. The change associated with allowing isolation of the affected main steam line is considered acceptable since manual isolation of the affected main steam line accomplishes the same action of the actuation instrumentation and operation with a main steam line isolated has been analyzed and shown to be acceptable as part of a safety evaluation. Some conditions may affect the isolation logic for only one main steam line. In these cases, it is not necessary to require a shutdown of the unit; rather, isolation of the affected line returns the system to a status where it can perform the remainder of its isolation function, and continued operation is allowed (although it may be at a reduced power level) consistent with the plant specific reload analyses which support operation with one main steam line out of service. The change is consistent with the ISTS.

L.4 CTS Tables 3.2.1 and 3.2.2 include requirements for Trip System Logics associated with the primary containment isolation instrumentation Trip Functions. These Trip Systems Logics are considered part of the primary containment isolation instrumentation Trip Functions and the requirements for these associated Trip System Logics to be operable are encompassed by the definition of operable. Therefore, the CTS Table 3.2.1 and 3.2.2 listing of Trip System Logics as separate Trip Functions is unnecessary and is deleted. With the deletion of separate Trip System Logic Trip Functions, the actions associated with inoperable Trip System Logic (CTS Table 3.2.1 Note 5 and CTS Table 3.2.2 Notes 2.A and 3) will now be governed by the actions for the individual proposed Table 3.2.2 primary containment isolation instrumentation Trip Functions. These proposed Table 3.2.2 Action Notes are less restrictive than the CTS Table 3.2.1 Note 5 and CTS Table 3.2.2 Notes 2.A and 3 actions. However, the proposed actions will ensure, in the event of inoperabilities, that consistent actions are applied to both primary containment isolation instrumentation Trip Functions and their associated Trip System Logics for the same level of degradation. This change is acceptable, since the allowed outage times of the proposed Table 3.2.2 Action Notes will limit operation to within the bounds of the applicable analysis, i.e., NEDC-31677-P-A, "Technical Specifications Improvement Analyses for BWR Isolation Actuation

No Significant Hazards Consideration

3.2.B and 4.2.B

Primary Containment Isolation

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, PROCEDURES, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents. The Bases, UFSAR, procedures, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the relocated details in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant reduction in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specifications, NUREG-1433, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.B/4.2.B - PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the requirements in Cold Shutdown or Refueling (with reactor coolant water temperature ≤ 212°F) for the Residual Heat Removal (RHR) Shutdown Cooling Isolation Trip Functions associated with RHR Shutdown Cooling isolation valves. This change will not result in any hardware or operating procedure changes. The RHR Shutdown Cooling isolation instrumentation is not assumed to be an initiator of any analyzed event. The role of the instrumentation is to protect that RHR System from overpressurization due to high reactor pressure. In Cold Shutdown or Refueling (with reactor coolant water temperature ≤ 212°F), the RHR Shutdown Cooling isolation instrumentation is not needed to protect that RHR System from overpressurization due to high reactor pressure. Therefore, this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change still ensures the affected instrumentation is required to be operable when it is necessary to perform its function. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change deletes the requirements in Cold Shutdown or Refueling (with reactor coolant water temperature ≤ 212°F) for the Residual Heat Removal (RHR) Shutdown Cooling Isolation Trip Functions associated with RHR Shutdown Cooling isolation valves. Due to the pressure and temperature limitations of Cold Shutdown or Refueling (with reactor coolant water temperature ≤ 212°F), the RHR Shutdown Cooling isolation instrumentation is not needed in these Modes to protect that RHR System from overpressurization due to high reactor pressure. As such, this change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will extend operation when one or more channels of the affected Primary Containment Isolation Trip Function instrumentation are inoperable and a loss of function has not occurred. The affected Primary Containment Isolation Trip Function instrumentation is not considered to be an initiator of any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Continued operation with inoperable Primary Containment Isolation Trip Function instrumentation channels will continue to be limited in accordance with Technical Specifications. Since the capability to perform the safety function will still be maintained in the proposed condition, the consequences of an accident occurring during the time allowed by proposed change are the same as the consequences currently allowed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change still ensures that continued operation in the applicable condition is not allowed when an associated Primary Containment Isolation Trip Function instrumentation channel is not capable of performing its required safety function. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change will extend operation when one or more channels of the affected Primary Containment Isolation Trip Function instrumentation are inoperable and a loss of function has not occurred. No change is being made in the manner in which systems relied upon in the safety analyses provide plant protection. The capability to perform the required safety function will still be maintained in the proposed condition. Plant safety margins continue to be maintained through the limitations established in the Technical Specifications. This change does not impact plant equipment design or operation, and there are no changes being made to safety limits or limiting safety system settings. The proposed change does not impact safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will allow continued operation with inoperable channels if the affected main steam line penetration is isolated. Isolated penetrations are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, isolating the penetration fulfills the post accident function of the isolation logic. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant (an evaluation has already been performed for the condition of one main steam line isolated). Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   When minimum conditions are not met for the Main Steam Line Isolation Trip Functions associated with Low-Low Reactor Vessel Water Level or Condenser Low Vacuum, the action is revised to allow isolation of the affected main steam line. Manual isolation of the affected main steam line accomplishes the same action of the actuation instrumentation and operation with a main steam line isolated has been analyzed and shown to be acceptable. The proposed change provides the benefit of avoiding a potential shutdown transient when the capability to satisfy the safety function of the affected instrumentation exists. Therefore, this change does not involve a significant reduction in a margin of safety since the required safety function of the inoperable channels will be fulfilled.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will relax the actions when one or more channels of primary containment isolation Trip Function instrumentation are inoperable due to inoperable Trip System Logic. The primary containment isolation Trip Function instrumentation is not considered to be an initiator of any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Continued operation with inoperable primary containment isolation Trip Function instrumentation channels will continue to be limited in accordance with Technical Specifications. Since the level of degradation allowed in the proposed actions are the same as the current actions, the consequences of an accident occurring during the time allowed by proposed change are the same as the consequences currently allowed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change still ensures that continued operation in the applicable condition is not allowed when an associated primary containment isolation Trip Function instrumentation channel is not capable of performing its required safety function. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change will relax actions when one or more channels of the affected primary containment isolation Trip Function instrumentation are inoperable. No change is being made in the manner in which systems relied upon in the safety analyses provide plant protection. Plant safety margins continue to be maintained through the limitations established in the Technical Specifications. This change does not impact plant equipment design or operation, and there are no changes being made to safety limits or limiting safety system settings. The proposed change does not impact safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.5 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change allows an isolated primary containment penetration to be opened under administrative controls. Primary containment isolation is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. Although primary containment isolation is considered in the mitigation of the consequences of an accident, the administrative controls provide acceptable compensatory actions to assure the penetration is isolated in the event of an accident. Therefore, the consequences of a previously analyzed event that may occur during the opening of the isolated line are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This change provides acceptable compensatory actions following failure of other equipment. The current requirements are based on providing a single active failure proof boundary to compensate for the loss of one of the two active isolation boundaries. This change provides an alternative which meets the original criteria of a single active failure proof boundary and is capable of returning the system to its original configuration (i.e., configuration which can provide a single active failure proof boundary.) Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

   The margin of safety considered in determining the required compensatory action is based on providing the single active failure proof boundary. Opening of primary containment penetrations on an intermittent basis is required for performing surveillances, repairs, routine evolutions, etc. which minimizes the possibility of a transient due to a required plant shutdown. The administrative controls provide a compensatory boundary in this condition. Since the compensatory boundary is capable of essentially meeting the same criteria of a single active failure proof boundary, the change does not involve a significant reduction in the margin of safety.

References

3.2.B and 4.2.B

Primary Containment Isolation

3.2.B/4.2.B REFERENCES
Primary Containment Isolation

1. Technical Requirements Manual.

2. UFSAR, Chapter 14.

3. UFSAR, Table 6.5.3.

4. UFSAR, Section 14.6.5.

5. UFSAR, Section 14.5.4.1.

6. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

7. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

Proposed

Technical Specifications

3.2.C and 4.2.C

Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

The reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation for each Trip Function in Table 3.2.3 shall be operable in accordance with Table 3.2.3.

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

1. The reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.3.

   When a channel is placed in an inoperable status solely for performance of required instrumentation surveillances, entry into the associated Limiting Conditions for Operation and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains reactor building ventilation isolation capability and Standby Gas Treatment System initiation capability.

2. Perform a Logic System Functional Test of reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation Trip Functions once every Operating Cycle.

Table 3.2.3 (page 1 of 1)
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1), (2)	2	Note 1	≥ 127.0 inches
2. High Drywell Pressure	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1)	2	Note 1	≤ 2.5 psig
3. High Reactor Building Ventilation Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1), (2), (3), (4)	1	Note 1	≤ 14 mR/hr
4. High Refueling Floor Zone Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL(1), (2), (3), (4)	1	Note 1	≤ 100 mR/hr

(1) With reactor coolant temperature > 212 °F.

(2) During operations with potential for draining the reactor vessel.

(3) During movement of irradiated fuel assemblies or fuel cask in secondary containment.

(4) During Alteration of the Reactor Core.

Table 3.2.3 ACTION Note

1. With one or more required Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation channels inoperable, take all of the applicable Actions in Notes 1.a and 1.b below.

   a. With one or more Trip Functions with one or more required channels inoperable:

      1) For Trip Functions 1 and 2, place inoperable channel in trip within 12 hours; and
      2) For Trip Functions 3 and 4, place inoperable channel in trip within 24 hours.

   b. With one or more Trip Functions with isolation or initiation capability not maintained:

      1) Restore isolation and initiation capability within 1 hour.

   If any applicable Action and associated completion time of Note 1.a or 1.b is not met, isolate the Reactor Building Ventilation System and place the Standby Gas Treatment System in operation within 1 hour.

Table 4.2.3 (page 1 of 1)
Reactor Building Ventilation Isolation and
Standby Gas Treatment System Initiation Instrumentation
Tests and Frequencies

TRIP FUNCTION	CHECK	FUNCTIONAL TEST	CALIBRATION
1. Low Reactor Vessel Water Level	NA	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
2. High Drywell Pressure	NA	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
3. High Reactor Building Ventilation Radiation	Once/Day	Every 3 Months	Every 3 Months
4. High Refueling Floor Zone Radiation	Once/Day During REFUELING	Every 3 Months	Every 3 Months

(1) Trip unit calibration only.

Proposed

Bases

3.2.C and 4.2.C

Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

BACKGROUND

The reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation automatically initiates closure of the Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs) and starts the Standby Gas Treatment (SGT) System. The function of these components and systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). Reactor Building (i.e., secondary containment) isolation and establishment of vacuum with the SGT System ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be operable, are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of reactor building ventilation isolation and Standby Gas Treatment System operation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a reactor building ventilation isolation and Standby Gas Treatment System initiation signal to the isolation and initation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation and initiation logic are (1) reactor vessel water level, (2) drywell pressure, (3) reactor building ventilation radiation, and (4) refueling floor zone radiation. Redundant sensor input signals from each parameter are provided for initiation and isolation.

For both the Low Reactor Vessel Water Level and High Drywell Pressure Trip Functions, the reactor building ventilation isolation and Standby Gas Treatment System initiation logic receives input from four channels. The outputs of the channels are arranged in one-out-of-two taken twice logics.

For the High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation Trip Functions, two radiation detectors and monitors are provided for each Trip Function. Each channel includes a radiation detector and associated monitor. The outputs of the channels are arranged in a one-out-of-two logic. In addition, the outputs of each channel are provided to both Trip Systems A and B. As such, any High Reactor Building Ventilation Radiation or High Refueling Floor Zone Radiation Trip Function channel will initiate reactor building ventilation isolation and Standby Gas Treatment System operation. (For the purposes of the Technical Specifications, the A radiation detectors and monitors should be considered to be associated with the Trip System A and the B radiation detectors and monitors should be considered to be associated with Trip System B.) Trip System A initiates startup of SGT subsystem A and initiates isolation of the reactor building supply and exhaust outboard isolation valves. Trip System B initiates startup of SGT subsystem B and initiates isolation of the reactor building supply and exhaust inboard isolation valves. As such, either Trip System isolates the secondary containment and provides the necessary filtration of fission products.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation and initiation signals generated by the reactor building ventilation isolation and Standby Gas Treatment System initiation

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

instrumentation are implicitly assumed in the safety analyses of References 2, 3, and 4, to initiate closure of the RBAVSIVs and start the SGT System to limit offsite doses.

Reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The operability of the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions specified in Table 3.2.3. Each Trip Function must have the required number of operable channels in each trip system, with their trip setpoints within the calculational as-found tolerances specified in plant procedures. Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.3. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions.

In general, the individual Trip Functions are required to be OPERABLE in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL (with reactor coolant temperature > 212°F), during operations with the potential for draining the reactor vessel (OPDRVs), during movement of irradiated fuel assemblies or fuel cask in secondary containment, and during Alteration of the Reactor Core; consistent with the Applicability for the SGT System and secondary containment requirements in Specifications 3.7.B and 3.7.C. Trip Functions that have different Applicabilities are discussed below in the individual Trip Functions discussion.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Trip Function by Trip Function basis.

1. Low Reactor Vessel Water Level

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite release. The Low Reactor Vessel Water Level Trip Function is one of the Trip Functions assumed to be operable and capable of providing isolation and initiation signals. The isolation and initiation of systems on Low Reactor Vessel Water Level support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Low Reactor Vessel Water Level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Low Reactor Vessel Water Level Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation and initiation function.

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low Reactor Vessel Water Level Trip Setting was chosen to be the same as the Reactor Protection System (RPS) Low Reactor Vessel Water Level Trip Setting (Specification 3.1.A), since this could indicate that the capability to cool the fuel is being threatened. The Trip Setting is referenced from the top of enriched fuel.

The Low Reactor Vessel Water Level Trip Function is required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL (with reactor coolant temperature > 212°F) where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. In COLD SHUTDOWN and REFUELING (with reactor coolant temperature ≤ 212°F), the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes; thus, this Trip Function is not required. In addition, the Trip Function is also required to be operable during OPDRVs to ensure that offsite dose limits are not exceeded if core damage occurs.

2. High Drywell Pressure

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite release. The isolation and initiation of systems on High Drywell Pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of High Drywell Pressure Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude performance of the isolation and initiation function.

The Trip Setting was chosen to be the same as the RPS High Drywell Pressure Trip Setting (Specification 3.1.A) since this is indicative of a loss of coolant accident (LOCA).

The High Drywell Pressure Trip Function is required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL (with reactor coolant temperature > 212°F) where considerable energy exists in the RCS; thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. This Trip Function is not required in COLD SHUTDOWN and REFUELING (with reactor coolant temperature ≤ 212°F) because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes.

3, 4. High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation

High reactor building ventilation radiation or refuel floor zone radiation is an indication of possible gross failure of the fuel cladding. The release may

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When High Reactor Building Ventilation Radiation or High Refueling Floor Zone Radiation is detected, secondary containment isolation and actuation of the SGT System are initiated to support actions to limit the release of fission products as assumed in the UFSAR safety analyses (Ref. 4).

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation signals are initiated from radiation detectors that are located on the ventilation exhaust duct coming from the reactor building and the refueling floor zones, respectively. Two channels of High Reactor Building Ventilation Radiation Trip Function and two channels of High Refueling Floor Radiation Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation and initiation function.

The Trip Settings are chosen to promptly detect gross failure of the fuel cladding.

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation Trip Functions are required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, REFUEL (with reactor coolant temperature > 212°F) where considerable energy exists in the RCS; thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. In COLD SHUTDOWN and REFUELING (with reactor coolant temperature ≤ 212°F), the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes; thus, these Trip Functions are not required. In addition, the Trip Functions are also required to be operable during OPDRVs, during movement of irradiated fuel assemblies or fuel cask in the secondary containment, and during Alteration of the Reactor Core, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

ACTIONS

Table 3.2.3 ACTION Note 1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours depending on the Trip Function (12 hours for those Trip Functions that have channel components common to RPS instrumentation, i.e., Trip Functions 1 and 2, and 24 hours for those Trip Functions that do not have channel components common to RPS instrumentation, i.e., all other Trip Functions), has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to operable status. This out of service time is only acceptable provided the associated Trip Function is still maintaining isolation capability (refer to next paragraph). If the inoperable channel cannot be restored to operable status within the allowable out of service time, the channel must be placed in the tripped condition per Table 3.2.3 Note 1.a.1) or 1.a.2), as applicable. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately,

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

ACTIONS (continued)

if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation or initiation), the Reactor Building Ventilation System must be isolated and the SGT System must be placed in operation within the next one hour. Isolating the Reactor Building Ventilation System and placing the SGT System in operation performs the intended function of the instrumentation and allows operation to continue.

Table 3.2.3 Note 1.b is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Trip Function result in a complete loss of isolation capability for the associated penetration flow path(s) or a complete loss of initiation capability for the SGT System. A Trip Function is considered to be maintaining isolation and initiation capability when sufficient channels are operable or in trip in both trip systems, such that a trip signal will be generated from the given Trip Function on a valid signal. This ensures that isolation of the two RBAVSIVs in the associated penetration flow path and the operation of the SGT System can be initiated on an isolation and initiation signal from the given Trip Function. For the Trip Functions 1 and 2, this would require each trip system to have one channel operable or in trip. For Trip Functions 3 and 4, this would require one channel to be operable or in trip. The one hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

If any applicable Action and associated Completion Time of Table 3.2.3 ACTION Note 1.a or 1.b are not met, the ability to isolate the secondary containment and start the SGT System cannot be ensured. Therefore, further actions must be performed to ensure the ability to maintain the secondary containment isolation and SGT System initiation function. Isolating the associated penetration flow path(s) and starting the associated SGT System within the next one hour performs the intended function of the instrumentation and allows operation to continue. One hour is sufficient for plant operations personnel to establish required plant conditions or to declare the associated components inoperable without unnecessarily challenging plant systems.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.C.1

As indicated in Surveillance Requirement 4.2.C.1, reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.3. Table 4.2.3 identifies, for each Trip Function, the applicable Surveillance Requirements.

Surveillance Requirement 4.2.C.1 also indicates that when a channel is placed in an inoperable status solely for performance of required instrumentation Surveillances, entry into associated LCO and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains isolation and initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

SURVEILLANCE REQUIREMENTS (continued)

applicable LCO entered and required Actions taken. This allowance is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RBAVSIVs will isolate the penetration flow path(s) and that the SGT System will initiate when necessary.

Surveillance Requirement 4.2.C.2

The Logic System Functional Test demonstrates the operability of the required initiation logic and simulated automatic operation for a specific channel. The testing required by the SGT System and RBAVSIVs Technical Specifications overlaps this Surveillance to provide testing of the assumed safety function. The Frequency of "once every Operating Cycle" is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the specified Frequency.

Table 4.2.3, Check

Performance of an Instrument Check once per day, for Trip Function 3, and once per day during REFUELING, for Trip Function 4, ensures that a gross failure of instrumentation has not occurred. An Instrument Check is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. An Instrument Check will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each Calibration. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit. The Frequency is based upon operating experience that demonstrates channel failure is rare. The Instrument Check supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

Table 4.2.3, Functional Test

For Trip Functions 1, 2, 3, and 4, a Functional Test is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of "Every 3 Months" is based on the reliability analysis of References 5 and 6.

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

SURVEILLANCE REQUIREMENTS (continued)

Table 4.2.3, Calibration

For Trip Functions 1, 2, 3, and 4, an Instrument Calibration is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. An Instrument Calibration leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The specified Instrument Calibration Frequencies are based upon the time interval assumptions for calibration used in the determination of the magnitude of equipment drift in the associated setpoint analyses.

For Trip Functions 1 and 2, a calibration of the trip units is required (Footnote (1)) once every 3 months. Calibration of the trip units provides a check of the actual setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the calculational as-found tolerances specified in plant procedures. The Frequency of every 3 months is based on the reliability analysis of Reference 6 and the time interval assumption for trip unit calibration used in the associated setpoint calculation.

REFERENCES

1. UFSAR, Section 5.3.

2. UFSAR, Section 7.17.2.

3. UFSAR, Section 14.6.3.6.

4. UFSAR, Section 14.6.4.4.

5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

Current

Technical Specifications

Markups

3.2.C and 4.2.C

Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the operational status of the plant instrumentation systems which initiate and control a protective function.

Objective:

To assure the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

When the system(s) it initiates or controls is required in accordance with Specification 3.5, the instrumentation which initiates the emergency core cooling system(s)shall be operable in accordance with Table 3.2.1.

B. Primary Containment Isolation

When primary containment integrity is required, in accordance with Specification 3.7, the instrumentation that initiates primary containment isolation shall be operable in accordance with Table 3.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

~~The instrumentation that initiates the isolation of the reactor building ventilation system and the actuation of the standby gas treatment system~~ shall be operable in accordance with Table 3.2.3.

The reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation for each Trip function in Table 3.2.3

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the surveillance requirements of the instrumentation systems which initiate and control a protective function.

Objective:

To verify the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.1.

B. Primary Containment Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.2.

{Move to separate pages}

1. The reactor building ventilation isolation and Standby Gas Treatment System initiation

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

Instrumentation ~~and logic systems~~ shall be ~~functionally~~ checked tested and calibrated as indicated in Table 4.2.3.

A.2

A.3

When a channel is placed in an inoperable status solely for performance of required instrumentation surveillances, entry into the associated Limiting Conditions for Operation and required actions may be delayed for up to 6 hours provided the associated Trip Function maintains reactor building ventilation isolation capability and Standby Gas Treatment System initiation capability.

34

A.4

A.1

INSTRUMENTATION

Actions When Required Channels are Inoperable

TABLE 3.2.3

REACTOR BUILDING VENTILATION ISOLATION & STANDBY GAS TREATMENT SYSTEM INITIATION

REQUIRED

	Minimum Number of ~~Operable~~ Instrument Channels per Trip System	Trip Function	Trip Setting	Required ACTION When ~~Minimum Conditions For Operation Are Not Satisfied~~
1.	2 (Notes ~~2~~, 3)	Low Reactor Vessel Water Level ~~(LT-2-3-57A/B(M), LT-2-3-58A/B(M))~~	~~Same as PCIS~~ ≥ 127.0 inches A.1	Note 1
2.	2 (Notes ~~2~~, 3)	High Drywell Pressure ~~(PT-5-12(A-D)(M))~~ LA.1	~~Same as PCIS~~ ≤ 2.5 psig A.1	Note 1
3.	1 (Notes ~~2~~, 3)	Reactor Building ~~Vent~~ VENTILATION Radiation High ~~(RM-17-452A/B)~~ A.6	≤14 m~~r~~R/hr	Note 1
4.	1 (Notes ~~2~~, 3)	Refueling Floor Zone Radiation ~~(RM-17-453A/B)~~ LA.1	≤100 m~~r~~R/hr	Note 1
	~~1~~	~~Reactor Building Vent Trip System Logic~~	~~--~~	~~Note 1~~
	~~1~~	~~Standby Gas Treatment Trip System Logic~~	~~--~~	~~Note 1~~ L.3

A.4 A.5

A.1

TABLE 3.2.3 ACTION NOTES

Action Note 1

1. If the minimum number of operable instrument channels is not available in either trip system, the reactor building ventilation system shall be isolated and the standby gas treatment system operated within 1 hour ~~until the instrumentation is repaired.~~ A.7 A.1 M.1 A.1

<Moved to proposed SR 4.2.B.1>

2. When a channel, ~~and/or the affected primary containment isolation valve~~, is placed in an inoperable status solely for performance of required instrumentation surveillances, entry into associated Limiting Conditions for operation and required ACTIONS may be delayed for up to 6 hours provided the associated Trip Function maintains Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation. A.4

3. Whenever Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation are required by Specification 3.7.B and 3.7.C, ~~there shall be two operable or tripped trip systems for each Trip Function~~, except as provided for below: A.8 LA.2 L.1 L.2

Action Note 1.b

a. With one or more automatic functions with isolation capability not maintained restore isolation and initiation capability in 1 hour ~~or take the ACTION required by Table 3.2.3.~~ A.7

Action Note 1.a

b. With one or more channels inoperable, place the inoperable channels'(s) in the tripped condition within:

1) 12 hours for trip functions common to RPS instrumentation, and
2) 24 hours for trip functions not common to RPS instrumentation,

or, initiate the ACTION required by Table 3.2.3. A.7

A.1

TABLE 4.2.3

~~MINIMUM~~ TESTS AND ~~CALIBRATION~~ FREQUENCIES

REACTOR BUILDING VENTILATION AND STANDBY GAS TREATMENT SYSTEM ISOLATION

	Trip Function	Functional Test (8) A.9	Calibration (8) A.9	Instrument Check
1.	Low Reactor Vessel Water Level	Every Three Months	Once/Operating Cycle — Every 3 Months (1) M.2	--
2.	High Drywell Pressure	Every Three Months	Once/Operating Cycle — Every 3 Months (1) M.2	--
3.	Reactor Building Ventilation Exhaust Radiation — High A.6	Every Three Months	Every Three Months	Once Each Day
4.	Refueling Floor Zone Radiation	Every Three Months	Every Three Months	Once Each Day During Refueling
SR 4.2.C.2	Reactor Building Vent Trip System Logic	Once/Operating Cycle	~~Once/Operating Cycle (Note 3)~~ A.10	-- A.2
SR 4.2.C.2	Standby Gas Treatment Trip System Logic	Once/Operating Cycle	~~Once/Operating Cycle (Note 3)~~ A.10	-- A.2

(1) Trip unit calibration only — M.2

A.1

TABLE 4.2 NOTES

1. Not used.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart. A.11

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system. A.10

4. This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. A.11

5. Deleted.

6. Deleted.

7. Deleted.

8. Functional tests and calibrations are not required when systems are not required to be operable. A.9

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

12. Trip system logic testing is not applicable to this function. If the required surveillance frequency (every Refueling Outage) is not met, functional testing of the Reactor Mode Switch-Shutdown Position function shall be initiated within 1 hour after the reactor mode switch is placed in Shutdown for the purpose of commencing a scheduled Refueling Outage.

13. Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).

A.11

Safety Assessment

Discussion of Changes

3.2.C and 4.2.C

Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

SAFETY ASSESSMENT OF CHANGES
TS: 3.2.C/4.2.C – REACTOR BUILDING VENTILATION ISOLATION
AND STANDBY GAS TREATMENT SYSTEM INITIATION INSTRUMENTATION

ADMINISTRATIVE

A.1 In the revision of the Vermont Yankee Nuclear Power Station (VYNPS) current Technical Specifications (CTS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the VYNPS Technical Specifications (TS) more consistent with human factor principles used in the Boiling Water Reactor Improved Standard Technical Specifications (ISTS), NUREG-1433, Rev. 2. These format and presentation changes are being made to improve usability and clarity. The changes are considered administrative.

A.2 CTS 4.2.C specifies that instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.3. In proposed Surveillance Requirement (SR) 4.2.C.1, the reference to "and logic system," is deleted since associated logic systems are considered part of the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation Trip Functions in both proposed and CTS Tables 3.2.3 and 4.2.3. It is not necessary to explicitly identify logic systems in CTS 4.2.C, since proposed SR 4.2.C.2 (CTS Table 4.2.3 requirements to perform Functional Tests of Trip System Logic) continues to require performance of surveillance testing of Trip System Logic (i.e., performance of Logic System Functional Tests for each reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation Trip Function). Therefore, this change is considered administrative.

A.3 CTS 4.2.C includes reference to CTS Table 4.2.3 for functional test and calibration requirements for reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation. CTS 4.2.C is revised, in proposed SR 4.2.C.1, to also include reference to check requirements consistent with CTS Table 4.2.3. This change is a presentation preference and does not alter the current requirements to periodically perform checks of certain reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation trip functions. Therefore, this change is considered administrative in nature.

A.4 CTS Table 3.2.3 Note 2 provides allowances to delay entry into actions for 6 hours for the situation of a channel inoperable solely for performance of surveillances. These allowances are moved to proposed SR 4.2.C.1. This change does not involve a technical change, but is only a difference of presentation preference. Therefore, this change is considered administrative.

A.5 Note 3 to CTS Table 3.2.3 provides actions when the minimum number of channels per trip system requirement is not met. These requirements are identified in a separate column in proposed Table 3.2.3 titled, "ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE." This change is a presentation preference and does not alter the current action requirements when required reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation channels are inoperable. Therefore, this change is considered administrative in nature.

ADMINISTRATIVE (continued)

A.6 CTS Table 3.2.3 includes a Reactor Building Vent Trip Function and a Refueling Floor Zone Radiation Trip Function. The purpose of these trip functions is to provide isolation of the reactor building and initiation of the Standby Gas Treatment System on receipt of a valid high radiation signal. As a result, the name of these trip functions are revised to "High Reactor Building Ventilation Radiation" and "High Refueling Floor Zone Radiation," respectively, in proposed Table 4.2.3 to more accurately reflect their functions. The design and operation of the actual instrumentation is unchanged. Therefore, this change is considered administrative.

A.7 CTS Table 3.2.3 Note 3 provides actions to be taken when one or more reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation channels are inoperable. In the event these actions are not completed within the specified time period, CTS Table 3.2.3, Notes 3.A and 3.B (as applicable) require the action required by Table 3.2.3 to be taken. For each of the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation trip functions, CTS Table 3.2.3 specifies that CTS Table 3.2.3 Note 1 is required. Therefore, as a human factors improvement, the actions of CTS Table 3.2.3 Notes 1 and 3 have been combined into one action (proposed Table 3.2.3 Action Note 1). Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.8 CTS Table 3.2.3 Note 3 specifies an Applicability for reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation of "When Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation are required by Specification 3.7.B and 3.7.C." Specifications 3.7.B and 3.7.C include the requirements for the Standby Gas Treatment System and the Secondary Containment System (which requires reactor building isolation). This change provides an explicit Applicability, in proposed Table 3.2.3 for each reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation trip function. The specified Applicabilities, in proposed Table 3.2.3, are consistent with the Modes and conditions when the Standby Gas Treatment System and the Secondary Containment System are required to be operable by Specifications 3.7.B and 3.7.C, respectively, except as provided and justified in changes L.2 and L.3 below. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. The change, providing explicit Mode or conditions of Applicability for each trip function, is consistent with the ISTS.

ADMINISTRATIVE (continued)

A.9 CTS Table 4.2 Note 8 states that functional tests and calibrations are not required when systems are not required to operable. The requirements of this Note are duplicated in the CTS definition 1.0.Z, "Surveillance Interval," which states that these tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but that these tests shall be performed on the instrument, component, or system prior to being required to be operable. Therefore, CTS Table 4.2 Note 8 is unnecessary and its deletion is considered to be administrative. The change is consistent with the ISTS.

A.10 For the Trip System Logic associated with the reactor building ventilation and the Standby Gas Treatment System instrumentation, CTS Table 4.2.3 includes requirements to perform a calibration of Trip System Logics once per Operating Cycle. These requirements are modified by Table 4.2 Note 3. Note 3 states, "Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system." The reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Functions of CTS Table 4.2.3 do not include any time delay relays or timers necessary for proper functioning of the trip systems. Therefore, this Note is deleted and, in proposed Table 4.2.3, the reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation (proposed Trip Functions 1 through 4) do not include calibration requirements for time delay relays or timers. As a result, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.11 CTS Table 4.2 Notes 2, 10, and 11 provide requirements that apply to ECCS instrumentation. The ECCS instrumentation is located in proposed Specifications 3.2.A and 4.2.A. Therefore, the requirements of CTS Table 4.2 Notes 2, 10, and 11 are physically moved and addressed in the changes for proposed Specifications 3.2.A and 4.2.A. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Notes 4, 12, and 13 provides requirements that apply to control rod block instrumentation. The control rod block instrumentation is located in proposed Specifications 3.2.E and 4.2.E. Therefore, the requirements of CTS Table 4.2 Notes 4, 12, and 13 are physically moved and addressed in the changes to proposed Specifications 3.2.E and 4.2.E. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Note 9 provides requirements that apply to post-accident monitoring instrumentation. The post-accident monitoring instrumentation is located in proposed Specifications 3.2.G and 4.2.G. Therefore, the requirements of CTS Table 4.2 Note 9 are physically moved and changes addressed in proposed Specifications 3.2.G and 4.2.G. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Table 3.2.3 Note 1 provides actions to isolate the Reactor Building Ventilation System and operate the Standby Gas Treatment System but does not state the time period in which these actions are to be completed. Proposed Table 3.2.3 Action Note 1 provides a one hour time period in which to complete the action of isolating the Reactor Building Ventilation System and placing the Standby Gas Treatment System in operation. Placing a limitation on the time allowed to complete the associated actions represents an additional restriction on plant operation since the time period allowed to complete the isolation will be controlled through TS. This change is consistent with the ISTS.

M.2 CTS Table 4.2.3 does not include explicit requirements to calibrate trip units. Proposed Table 4.2.3 requires calibration of the trip units of the following Trip Functions every 3 months: Low Reactor Vessel Water Level (proposed Table 4.2.3 Trip Function 1) and High Drywell Pressure (proposed Table 4.2.3 Trip Function 2). The trip units of these Trip Functions are currently required by CTS Table 4.2.3 to be calibrated with the rest of the associated instrument loops once per operating cycle. Therefore, this change is more restrictive. This change is necessary to ensure consistency with assumptions regarding trip unit calibration frequency used in the associated setpoint calculations. This change is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 3.2.3 details relating to system design and operation (i.e., the specific instrument tag numbers) are unnecessary in the TS and are proposed to be relocated to the Technical Requirements Manual (TRM). Proposed Specification 3.2.C and Table 3.2.3 require the reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation Trip Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.3 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Functions are maintained operable. As such, the relocated details are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TRM are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.2 CTS Table 3.2.3 Note 3 contains design and operational details of the reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation (i.e., there shall be two operable or tripped trip systems for each Trip Function). These details are not necessary to ensure the operability of associated reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation. Therefore, the information in this note is to be relocated to Specification 3.2.C Bases and reference to this information is deleted from VYNPS TS. The requirements of Specification 3.2.C and the

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (continued) associated Surveillance Requirements for the associated reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation are adequate to ensure the instruments are maintained operable. As such, these relocated requirements are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TS Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

"Specific"

L.1 The Applicability of the Low Reactor Vessel Water Level Trip Function of CTS Table 3.2.3 is "When Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation are required by Specification 3.7.B and 3.7.C." The requirements of Specifications 3.7.B and 3.7.C, and as a result the requirements for this Trip Function, are applicable in the Run, Startup/Hot Standby, Hot Shutdown, and Refuel (with reactor coolant water temperature > 212°F) Modes and during operations with a potential for draining the reactor vessel, during movement of irradiated fuel assemblies or fuel cask in secondary containment, and during alteration of the reactor core. The Low Reactor Vessel Water Level Trip Function is required to support the operability of the Secondary Containment System and the Standby Gas Treatment System to ensure fission products entrapped within secondary containment are treated prior to discharge to the environment. When the plant is in Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F), the probability and consequences of a design basis accident that is postulated to leak fission products into secondary containment are reduced due to the temperature and pressure limitations in these Modes and conditions. However, in Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F), activities are conducted for which significant releases of radioactivity are postulated due to reductions in reactor vessel water level. As a result, the Low Reactor Vessel Water Level Trip Function is required to be operable in Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F), when activities are in progress which could result in reactor vessel water level reductions (i.e., during operations with the potential for draining the reactor vessel (OPDRVs)). Low Reactor Vessel Water Level actuation of the reactor building ventilation isolation and the Standby Gas Treatment System initiation is not assumed to mitigate the consequences of postulated events in the Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F) Mode when OPDRVs are not being conducted. In the Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F) Mode, when other activities are conducted for which significant releases of radioactivity are postulated and OPDRVs are not being conducted, other reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Functions are required to be operable to generate the required isolation and initiation signals if required. As such, proposed Table 3.2.3 Trip Function 2 is revised to only be required in the Run, Startup/Hot Standby, Hot Shutdown and Refuel (with reactor coolant temperature > 212°F) Modes and during OPDRVs. This change is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 The Applicability of the High Drywell Trip Function of CTS Table 3.2.3 is "When Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation are required by Specification 3.7.B and 3.7.C." The requirements of Specifications 3.7.B and 3.7.C, and as a result the requirements for this Trip Function, are applicable in the Run, Startup/Hot Standby, Hot Shutdown, and Refuel (with reactor coolant water temperature > 212°F) Modes and during operations with a potential for draining the reactor vessel, during movement of irradiated fuel assemblies or fuel cask in secondary containment, and during alteration of the reactor core. The High Drywell Pressure Trip Function is required to support the operability of the Secondary Containment System and the Standby Gas Treatment System to ensure fission products entrapped within secondary containment are treated prior to discharge to the environment. When the plant is in Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F), the probability and consequences of a design basis accident that is postulated to leak fission products into secondary containment are reduced due to the temperature and pressure limitations in these Modes and conditions. In addition, in these Modes or conditions, there is insufficient energy in the reactor vessel to pressurize the primary containment and the primary containment is not required to be operable. As a result, High Drywell Pressure actuation of the reactor building ventilation isolation and the Standby Gas Treatment System initiation is not assumed to mitigate the consequences of postulated events in the Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F) Mode. In the Cold Shutdown or Refuel (with reactor coolant water temperature ≤ 212°F) Mode, when other activities are conducted for which significant releases of radioactivity are postulated, other reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Functions are required to be operable to generate the required isolation and initiation signals if required. As such, proposed Table 3.2.3 Trip Function 2 is revised to only be required in the Run, Startup/Hot Standby, Hot Shutdown and Refuel (with reactor coolant temperature > 212°F) Modes. This change is consistent with the ISTS.

L.3 CTS Table 3.2.3 includes requirements for Trip System Logics associated with the reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation Trip Functions. These Trip Systems Logics are considered part of the reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation Trip Functions and the requirements for these associated Trip System Logics to be operable are encompassed by the definition of operable. Therefore, the CTS Table 3.2.3 listing of Trip System Logics as separate Trip Functions is unnecessary and is deleted. With the deletion of separate Trip System Logic Trip Functions, the actions associated with inoperable Trip System Logic (CTS Table 3.2.3 Note 1) will now be governed by the actions for the individual proposed Table 3.2.3 reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation Trip Functions. These proposed Table 3.2.3 Action Notes are less restrictive than the CTS Table 3.2.3 Note 1 actions. However, the proposed actions will ensure, in the event of inoperabilities,

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (continued) that consistent actions are applied to both reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation Trip Functions and their associated Trip System Logics for the same level of degradation. This change is acceptable, since the allowed outage times of the proposed Table 3.2.3 Action Notes will limit operation to within the bounds of the applicable analysis, i.e., NEDC-31677-P-A, "Technical Specifications Improvement Analyses for BWR Isolation Actuation Instrumentation, Part 2," July 1990, and NEDC-30851-P-A Supplement 2, "Technical Specifications Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989. Application of these analyses to the VYNPS reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation Trip Functions, including the associated Trip System Logics, was approved by the NRC in VYNPS License Amendment No. 186 dated April 3, 2000. This change is consistent with the ISTS.

RELOCATED SPECIFICATIONS

None

No Significant Hazards Consideration

3.2.C and 4.2.C

Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, PROCEDURES, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents. The Bases, UFSAR, procedures, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the relocated details in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant reduction in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specifications, NUREG-1433, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.C/4.2.C. – REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INIATION INSTRUMENTATION

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will eliminate the Current Technical Specification 3.2.C Applicability requirements of the Low Reactor Vessel Water Level Trip Function for Modes or conditions other than Run, Startup/Hot Standby, Hot Shutdown, Refuel (with reactor coolant temperature > 212°F) and during operations with the potential for draining the reactor vessel (OPDRVs). This reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function is not assumed to be an initiator of any analyzed accident. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, this Trip Function is not credited for mitigation of any accident or transient in Modes or conditions other than Run, Startup/Hot Standby, Hot Shutdown, Refuel (with reactor coolant temperature > 212°F) and during OPDRVs. As such, the consequences of an accident occurring with the proposed change are the same as the consequences of an accident occurring with current requirements. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change still ensures the affected instrumentation is required to be operable when it is necessary to perform its function. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change is acceptable because it does not impact the ability of the affected reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation to perform its intended function which is to support the Secondary Containment System and the Standby Gas Treatment System in the performance of their safety functions. In addition, the change is consistent with the safety analysis since the reactor building ventilation isolation and the Standby Gas Treatment System initiation on low reactor vessel water level is a not assumed to mitigate postulated events in Modes or conditions other than Run, Startup/Hot Standby, Hot Shutdown, Refuel (with reactor coolant temperature > 212°F) and during OPDRVs. Since the change has no effect on any safety analysis assumptions or initial conditions, the margins of safety continue to be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.C/4.2.C. – REACTOR BUILDING VENTILATION ISOLATION
AND STANDBY GAS TREATMENT SYSTEM INIATION INSTRUMENTATION

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will eliminate the Current Technical Specification 3.2.C Applicability requirements of the High Drywell Pressure Trip Function for Modes or conditions other than Run, Startup/Hot Standby, Hot Shutdown and Refuel (with reactor coolant temperature > 212°F). This reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function is not assumed to be an initiator of any analyzed accident. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, this Trip Function is not credited for mitigation of any accident or transient in Modes or conditions other than Run, Startup/Hot Standby, Hot Shutdown and Refuel (with reactor coolant temperature > 212°F). As such, the consequences of an accident occurring with the proposed change are the same as the consequences of an accident occurring with current requirements. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change continues to ensure the affected instrumentation is capable of performing its function as assumed in the safety analyses. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change is acceptable because it does not impact the ability of the affected reactor building ventilation isolation and the Standby Gas Treatment System initiation instrumentation to perform its intended function which is to support the Secondary Containment System and the Standby Gas Treatment System in the performance of their safety functions. In addition, the change is consistent with the safety analysis since reactor building ventilation isolation and the Standby Gas Treatment System initiation on high drywell pressure is a not assumed to mitigate postulated events in Modes or conditions other than Run, Startup/Hot Standby, Hot Shutdown and Refuel (with reactor coolant temperature > 212°F). Since the change has no effect on any safety analysis assumptions or initial conditions, the margins of safety continue to be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.C/4.2.C. – REACTOR BUILDING VENTILATION ISOLATION
AND STANDBY GAS TREATMENT SYSTEM INIATION INSTRUMENTATION

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will relax the actions when one or more channels of reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function instrumentation are inoperable due to inoperable Trip System Logic. The reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function instrumentation is not considered to be an initiator of any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Continued operation with inoperable reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function instrumentation channels will continue to be limited in accordance with Technical Specifications. Since the level of degradation allowed in the proposed actions is the same as the current actions, the consequences of an accident occurring during the time allowed by proposed change are the same as the consequences currently allowed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change still ensures that continued operation in the applicable condition is not allowed when an associated reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function instrumentation channel is not capable of performing its required safety function. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change will relax actions when one or more channels of the affected reactor building ventilation isolation and the Standby Gas Treatment System initiation Trip Function instrumentation are inoperable. No change is being made in the manner in which systems relied upon in the safety analyses provide plant protection. Plant safety margins continue to be maintained through the limitations established in the Technical Specifications. This change does not impact plant equipment design or operation, and there are no changes being made to safety limits or limiting safety system settings. The proposed change does not impact safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

References

3.2.C and 4.2.C

Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

3.2.C/4.2.C REFERENCES
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

1. UFSAR, Section 5.3.

2. UFSAR, Section 7.17.2.

3. UFSAR, Section 14.6.3.6.

4. UFSAR, Section 14.6.4.4.

5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

Proposed

Technical Specifications

3.2.D and 4.2.D

Off-Gas System Isolation

3.2 LIMITING CONDITIONS FOR OPERATION

D. Deleted.

E. Control Rod Block Actuation

The control rod block instrumentation for each Trip Function in Table 3.2.5 shall be operable in accordance with Table 3.2.5.

4.2 SURVEILLANCE REQUIREMENTS

D. Deleted.

E. Control Rod Block Actuation

1. The control rod block instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.5.

   When a Rod Block Monitor channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains control rod block initiation capability.

Current

Technical Specifications

Markups

3.2.D and 4.2.D

Off-Gas System Isolation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

D. ~~Off-Gas System Isolation~~ R.1

Deleted.

~~During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.~~

E. Control Rod Block Actuation

During reactor power operation the instrumentation that initiates control rod block shall be operable in accordance with Table 3.2.5.

F. Mechanical Vacuum Pump Isolation Instrumentation

When the reactor is in the RUN or STARTUP/HOT STANDBY Mode and the mechanical vacuum pump is in service, four (4) channels of the High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be operable, except as provided below.

1. With one or more channels inoperable, within 12 hours:

   a. Restore the inoperable channel(s) to operable status; or

   b. Place the inoperable channel(s) or associated trip system in the trip condition (not applicable if the inoperable channel is the result of an inoperable mechanical vacuum pump isolation valve).

4.2 SURVEILLANCE REQUIREMENTS

D. ~~Off-Gas System Isolation~~ R.1

Deleted.

~~Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.~~

<MOVE TO SEPARATE PAGE>

E. Control Rod Block Actuation A.2

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5.

F. Mechanical Vacuum Pump Isolation Instrumentation

The High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be checked, functionally tested and calibrated as indicated in Surveillance Requirements 4.2.F.1, 2, 3, 4 and 5.

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required actions may be delayed for up to six (6) hours provided the associated trip function maintains mechanical vacuum pump isolation capability.

1. Perform an instrument check once each day.

2. Perform an instrument functional test once every three (3) months.

<MOVE TO SEPARATE PAGE>

TABLE 3.2.4

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
1	Time Delay (Stack Off-Gas Valve Isolation) (15TD & 16TD)	≤ 2 minutes ≤ 30 minutes	Note 1
1	Trip System Logic	--	Note 1

Note 1 - At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

R.1

VYNPS

TABLE 4.2.4

MINIMUM TEST AND CALIBRATION FREQUENCIES

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Augmented Off-Gas Trip System Logic (AOG)	Once/Operating Cycle (Note 2)	Once/Operating Cycle (Note 3)	--

R.1

A.1

TABLE 4.2 NOTES

1. Not used.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart. R.1

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.

4. This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. A.2

5. Deleted.

6. Deleted.

7. Deleted.

8. Functional tests and calibrations are not required when systems are not required to be operable R.1

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

12. Trip system logic testing is not applicable to this function. If the required surveillance frequency (every Refueling Outage) is not met, functional testing of the Reactor Mode Switch-Shutdown Position function shall be initiated within 1 hour after the reactor mode switch is placed in Shutdown for the purpose of commencing a scheduled Refueling Outage.

13. Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).

A.2

Safety Assessment

Discussion of Changes

3.2.D and 4.2.D

Off-Gas System Isolation

SAFETY ASSESSMENT OF CHANGES
CTS: 3.2.D/4.2.D – OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

ADMINISTRATIVE

A.1 In the revision of the Vermont Yankee Nuclear Power Station (VYNPS) current Technical Specifications (CTS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the VYNPS Technical Specifications (TS) more consistent with human factor principles used in the Boiling Water Reactor Improved Standard Technical Specifications (ISTS), NUREG-1433, Rev. 2. These format and presentation changes are being made to improve usability and clarity. The changes are considered administrative.

A.2 CTS 3.2.E, 4.2.E, and Table 4.2 Notes 4, 12, and 13 provide requirements that apply to control rod block instrumentation. The control rod block instrumentation is located in proposed Specifications 3.2.E and 4.2.E. Therefore, the requirements of CTS 3.2.E, 4.2.E, and Table 4.2 Notes 4, 12, and 13 are physically moved and addressed in the changes to proposed Specifications 3.2.E and 4.2.E. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Note 9 provides requirements that apply to post-accident monitoring instrumentation. The post-accident monitoring instrumentation is located in proposed Specifications 3.2.G and 4.2.G. Therefore, the requirements of CTS Table 4.2 Note 9 are physically moved and changes addressed in proposed Specifications 3.2.G and 4.2.G. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Notes 10 and 11 provide requirements that apply to ECCS instrumentation. The ECCS instrumentation is located in proposed Specifications 3.2.A and 4.2.A. Therefore, the requirements of CTS Table 4.2 Notes 10 and 11 are physically moved and addressed in the changes for proposed Specifications 3.2.A and 4.2.A. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 3.2.D/4.2.D OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

LCO Statement:

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

Discussion:

The radioactive off-gas processing system is not a safety system and is not connected to the primary coolant piping. The off-gas isolation instrumentation is used to ensure conformance with the discharge limits of 10 CFR 20. There is another Specification (Specification 3.8.K/4.8.K, Steam Jet Air Ejector) that ensures 10 CFR 100 limits are not exceeded in the event of a failure of the radioactive off-gas processing system. Information provided by this instrumentation on radiation levels would be limited or no use in identifying/assessing core damage in the event of an accident and they are not installed to detect excessive reactor coolant leakage.

Comparison to Screening Criteria:

1. Off-gas system isolation instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a Design Basis Accident (DBA).

2. Although off-gas activity is an initial condition of a DBA, this process variable is addressed by another Technical Specification. Criterion 2 is satisfied for the process variable (steam jet air ejector radioactivity) that is addressed by another Technical Specification. However, Criterion 2 is not satisfied for off-gas system isolation instrumentation, since this instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Off-gas system isolation instrumentation is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 145) of NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," dated November 1987, the loss of the off-gas system isolation instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. VYNPS has reviewed this evaluation, considers it applicable to VYNPS, and concurs with the assessment.

RELOCATED SPECIFICATIONS

R.1
(continued)

Conclusion:

Since the screening criteria have not been satisfied, the Off-Gas System Isolation Instrumentation LCO, Actions, and Surveillances will be relocated to the Technical Requirements Manual. Changes to the Technical Requirements Manual are controlled using 10 CFR 50.59.

No Significant Hazards Consideration

3.2.D and 4.2.D

Off-Gas System Isolation

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36 (c)(2)(ii). The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no significant reduction in a margin of safety will be permitted.

   The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specification, NUREG-1433 approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3.2.D/4.2.D – OFF – GAS ISOLATION INSTRUMENTATION

There were no specific less restrictive changes identified for this Specification.

Proposed

Technical Specifications

3.2.E and 4.2.E

Control Rod Block Actuation

And related Technical Specifications:

3.3 – Control Rod System

3.11 – Reactor Fuel Assemblies

3.2 LIMITING CONDITIONS FOR OPERATION

D. Deleted.

E. Control Rod Block Actuation

The control rod block instrumentation for each Trip Function in Table 3.2.5 shall be operable in accordance with Table 3.2.5.

4.2 SURVEILLANCE REQUIREMENTS

D. Deleted.

E. Control Rod Block Actuation

1. The control rod block instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.5.

   When a Rod Block Monitor channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains control rod block initiation capability.

Table 3.2.5 (page 1 of 1)
Control Rod Block Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP FUNCTION	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Rod Block Monitor				
a. Upscale (Flow Bias)	> 30% RATED THERMAL POWER	2	Note 1	$\leq 0.66(W)+N^{(2)}$ with a maximum as defined in the COLR
b. Downscale	> 30% RATED THERMAL POWER	2	Note 1	≥ 2/125 full scale
c. Inop	> 30% RATED THERMAL POWER	2	Note 1	NA
2. Reactor Mode Switch - Shutdown Position	(1)	2	Note 2	NA

(1) When reactor mode switch is in the shutdown position.

(2) Trip Setting $\leq 0.66\ (W-\Delta W)+N$ for single loop operation.

Table 3.2.5 ACTION Notes

1. With one or two RBM channels inoperable, take all of the applicable Actions in Notes 1.a and 1.b below.

   a. If one RBM channel is inoperable, restore the inoperable channel to operable status within 24 hours.

   b. If the required Action and associated completion time of Note 1.a above is not met, or if two RBM channels are inoperable, place one RBM channel in trip within the next hour.

2. With one or more Reactor Mode Switch - Shutdown Position channels inoperable, immediately suspend control rod withdrawal and immediately initiate Actions to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

Table 4.2.5 (page 1 of 1)
Control Rod Block Instrumentation
Tests and Frequencies

TRIP FUNCTION	FUNCTIONAL TEST	CALIBRATION
1. Rod Block Monitor (RBM)		
a. Upscale (Flow Bias)	Every 3 Months	Every 3 Months[(2)]
b. Downscale	Every 3 Months	Every 3 Months[(2)]
c. Inop	Every 3 Months	NA
2. Reactor Mode Switch - Shutdown Position	Every Refueling Outage[(1)]	NA

(1) Not required to be performed until 1 hour after the reactor mode switch is placed in the shutdown position.

(2) Neutron detectors are excluded.

3.3 LIMITING CONDITIONS FOR OPERATION

pressure, are fully inserted, no more than two rods may be moved.

4. Control rod patterns and the sequence of withdrawal or insertion shall be established such that the rod drop accident limit of 280 cal/g is not exceeded.

5. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate greater than or equal to three counts per second.

6. Deleted.

4.3 SURVEILLANCE REQUIREMENTS

(c) Out-of-sequence control rods in each distinct RWM group shall be selected and the annunciator of the selection errors verified.

(d) An out-of-sequence control rod shall be withdrawn no more than three notches and the rod block function verified.

4. The control rod pattern and sequence of withdrawal or insertion shall be verified to comply with Specification 3.3.B.4.

5. Prior to control rod withdrawal for startup or during refueling, verification shall be made that at least two source range channels have an observed count rate of at least three counts per second.

6. Deleted.

3.3 LIMITING CONDITIONS FOR OPERATION

C. Scram Insertion Times

1.1 The average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.358
36	25.34	0.912
26	46.18	1.468
06	87.84	2.686

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.379
36	25.34	0.967
26	46.18	1.556
06	87.84	2.848

4.3 SURVEILLANCE REQUIREMENTS

7. The scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing under administrative control.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power with reactor pressure above 800 psig all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. During or following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the start up test report.

3.11 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

1. During operation at $\geq$25% Rated Thermal Power the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. For core flows other than rated, the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is provided in the Core Operating Limits Report. If at any time during operation at $\geq$25% Rated Thermal Power it is determined by normal surveillance that the limiting value for MCPR is being exceeded, MCPR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor power shall be brought to <25% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined once within 12 hours after $\geq$25% Rated Thermal Power, daily during operation at $\geq$25% Rated Thermal Power thereafter.

Proposed

Bases

3.2.E and 4.2.E

Control Rod Block Actuation

And related Technical Specification Bases:

3.3 – Control Rod System

3.11 – Reactor Fuel Assemblies

BASES: 3.2.E/4.2.E CONTROL ROD BLOCK ACTUATION

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During shutdown conditions, control rod blocks from the Reactor Mode Switch—Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM (Ref.1) is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a Fuel Cladding Integrity Safety Limit violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the 30% RATED THERMAL POWER setpoint when a non-peripheral control rod (except control rod 35-34) is selected. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals. One RBM channel averages the signals from LPRM detectors at the A and C positions in the assigned LPRM assemblies, while the other RBM channel averages the signals from LPRM detectors at the B and D positions. Assignment of LPRMs to be used in RBM averaging is controlled by the selection of control rods. The RBM is automatically bypassed and the output set to zero if a peripheral rod (or control rod 35-34) is selected or the APRM used to normalize the RBM reading is at < 30% RATED THERMAL POWER. If any LPRM detector assigned to an RBM is bypassed, the computed average signal is automatically adjusted to compensate for the number of LPRM input signals. The minimum number of LPRM inputs required for each RBM channel to prevent an instrument inoperative trip is four when using four LPRM strings, three when using three LPRM strings, and two when using two LPRM strings. Each RBM also receives a recirculation loop flow signal from the associated flow converter.

When a control rod is selected, the gain of each RBM channel output is normalized to a reference APRM. The gain setting is held constant during the movement of that particular control rod to provide an indication of the change in the relative local power level. If the indicated power increases above the preset limit, a rod block will occur. In addition, to preclude rod movement with an inoperable RBM, a downscale trip and an inoperable trip are provided.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained (Ref. 2). This Trip Function prevents inadvertent criticality as the result of a control rod withdrawal during COLD SHUTDOWN and HOT SHUTDOWN or during a Refueling Outage when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

BASES: 3.2.E/4.2.E CONTROL ROD BLOCK ACTUATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 1.b, 1.c Rod Block Monitor

The RBM is designed to prevent violation of the Fuel Cladding Integrity Safety Limit and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. The cycle-specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is operating at rated power with a control rod pattern that results in the core being placed on thermal design limits. The condition is analyzed to ensure that the results obtained are conservative; the approach also serves to demonstrate the functions of the RBM.

The RBM Trip Functions satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels of each of the RBM Trip Functions are required to be operable, with their trip setpoints within the calculational as-found tolerances specified in plant procedures, as applicable, to ensure that no single instrument failure can preclude a rod block from these Trip Functions. In addition, to provide adequate coverage of the entire core, LPRM inputs for each RBM channel are required from greater than or equal to half the total number of inputs from any LPRM level. The upper limit of the RBM Upscale (Flow Bias) Trip Function is clamped to provide protection at greater than 100% rated core flow. This clamped value is cycle-specific and is included in the Core Operating Limits Report. Trip Settings are specified for RBM Upscale (Flow Bias) and RBM Downscale Trip Functions. The terms for the Trip Setting of the RBM Upscale (Flow Bias) Trip Function are defined as follows: W is percent of rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 X 10^6 lbs/hr core flow; and ΔW is the difference between two loop and single loop drive flow a the same core flow (this difference must be accounted for during single loop operation). $\Delta W = 0$ for two loop operation.

Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.5. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions.

The RBM is assumed to mitigate the consequences of an RWE event when operating > 30% RATED POWER THERMAL and a non-peripheral control rod is selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the Fuel Cladding Integrity Safety Limit and, therefore, the RBM is not required to be operable.

BASES: 3.2.E/4.2.E CONTROL ROD BLOCK ACTUATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Reactor Mode Switch - Shutdown Position

During HOT SHUTDOWN and COLD SHUTDOWN, and during Refueling Outages when the reactor mode switch is in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch-Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch-Shutdown Position Trip Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels are required to be operable to ensure that no single channel failure will preclude a rod block when required. There is no Trip Setting for this Trip Function since the channels are mechanically actuated based solely on reactor mode switch position. During shutdown conditions (HOT SHUTDOWN and COLD SHUTDOWN, and Refueling Outages when the reactor mode switch is in the shutdown position), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be operable. With the reactor mode switch in the refueling position, the refuel position one-rod-out interlock provides the required control rod withdrawal blocks.

ACTIONS

Table 3.2.5 ACTION Note 1

With one RBM Trip Function 1.a, 1.b, or 1.c channel inoperable, the remaining operable channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining operable RBM channel can result in no control rod block capability for the RBM. For this reason, Table 3.2.5 ACTION Note 1.a requires restoration of the inoperable channel to operable status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining operable channel.

If the Table 3.2.5 ACTION Note 1.a required action is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM Trip Function 1.a, 1.b, or 1.c channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

Table 3.2.5 ACTION Note 2

With one Reactor Mode Switch-Shutdown Position control rod withdrawal block channel inoperable, the remaining operable channel is adequate to perform the

BASES: 3.2.E/4.2.E CONTROL ROD BLOCK ACTUATION

ACTIONS (continued)

control rod withdrawal block function. However, since the required actions of Table 3.2.5 ACTION Note 2 are consistent with the normal action of an operable Reactor Mode Switch–Shutdown Position Trip Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate Shutdown Margin ensured by Specification 3.3.A.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.E.1

As indicated in Surveillance Requirement 4.2.E.1, control rod block instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.5. Table 4.2.5 identifies, for each Trip Function, the applicable Surveillance Requirements.

Surveillance Requirement 4.2.E.1 also indicates that when an RBM channel is placed in an inoperable status solely for performance of required instrumentation Surveillances, entry into associated LCO and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable LCO entered and required Actions taken. This allowance is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

Table 4.2.5, Functional Test

For Trip Functions 1.a, 1.b, and 1.c, a Functional Test is performed on each required channel to ensure that the channel will perform the intended function. The Functional Test of the RBM channels includes the Reactor Manual Control "Select Relay Matrix" System input. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of "Every 3 Months" is based on the reliability analysis of Reference 5.

For Trip Function 2, a Functional Test is performed to ensure that the entire channel will perform the intended function. The Functional Test for the Reactor Mode Switch–Shutdown Position Trip Function is performed by attempting to withdraw any control rod with the reactor mode switch in the

BASES: 3.2.E/4.2.E CONTROL ROD BLOCK ACTUATION

SURVEILLANCE REQUIREMENTS (continued)

shutdown position and verifying a control rod block occurs. As noted in Table 4.2.5 Footnote (1), the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into the HOT SHUTDOWN and COLD SHUTDOWN Modes if the "Every Refueling Outage" Frequency is not met. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the Surveillance Requirement. The Frequency of "Every Refueling Outage" is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the specified Frequency.

Table 4.2.5, Calibration

For Trip Functions 1.a and 1.b, an Instrument Calibration is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. An Instrument Calibration leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The specified Instrument Calibration Frequencies are based upon the time interval assumptions for calibration used in the determination of the magnitude of equipment drift in the associated setpoint analyses.

The RBM is automatically bypassed when power is below a specified value or if a peripheral control rod is selected. The power level is determined from the APRM signals input to each RBM channel. The automatic bypass setpoint must be verified periodically to be $\leq$ 30% RATED THERMAL POWER. As a result, the Instrument Calibration of Trip Function 1.a must also include calibration of the RBM Reference Downscale function (i.e., RBM Upscale (Flow Bias) Trip Function is not bypassed when > 30% RATED THERMAL POWER). In addition, it must also be verified that the RBM is not bypassed when a control rod that is not a peripheral control rod is selected (only one non-peripheral control rod is required to be verified). If any bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the APRM channel can be placed in the conservative condition to enable the RBM. If placed in this condition, the Surveillance Requirement is met and the RBM channel is not considered inoperable.

As noted in Footnote (2), neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day heat balance calibration and the 2000 MWD/T LPRM calibration against the Traversing Incore Probe System of the Reactor Protection System Technical Specification.

BASES: 3.2.E/4.2.E CONTROL ROD BLOCK ACTUATION

REFERENCES

1. UFSAR, Section 7.5.8.

2. UFSAR, Section 7.7.4.3.2.

3. UFSAR, Section 14.5.3.1.

4. GENE-770-06-1-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

5. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system provides a scram function in noncoincident configuration. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the continuous withdrawal of a designated single control rod could result in a violation of the MCPR safety limit or the 1% plastic strain limit.

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily when operating at ≥25% Rated Thermal Power to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. The 12 hour allowance after thermal power ≥25% Rated Thermal Power is achieved is acceptable given the large inherent margin to operating limits at low power levels.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

If a reactor power distribution limit is exceeded, an assumption regarding an initial condition of the DBA analysis, transient analyses, or the fuel design analysis may not be met. Therefore, prompt action should be taken to restore the APLHGR, LHGR or MCPR to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour completion time is sufficient to restore the APLHGR, LHGR, or MCPR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR, LHGR, or MCPR out of specification.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

Current

Technical Specifications

Markups

3.2.E and 4.2.E

Control Rod Block Actuation

And related Technical Specifications:

3.3 – Control Rod System

3.11 – Reactor Fuel Assemblies

A.1

3.2 LIMITING CONDITIONS FOR OPERATION	4.2 SURVEILLANCE REQUIREMENTS

D. Off-Gas System Isolation (3.2)

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

D. Off-Gas System Isolation (4.2)

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.

A.2

E. Control Rod Block Actuation (3.2)

A.3

~~During reactor power operation, the instrumentation that initiates control rod block~~ THE CONTROL ROD BLOCK INSTRUMENTATION FOR EACH TRIP FUNCTION IN TABLE 3.2.5 shall be operable in accordance with Table 3.2.5.

E. Control Rod Block Actuation (4.2)

~~Instrumentation and logic systems~~ 1. THE CONTROL ROD BLOCK shall be functionally tested and calibrated as indicated in Table 4.2.5.

A.4

F. Mechanical Vacuum Pump Isolation Instrumentation (3.2)

When the reactor is in the RUN or STARTUP/HOT STANDBY Mode and the mechanical vacuum pump is in service, four (4) channels of the High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be operable, except as provided below.

1. With one or more channels inoperable, within 12 hours:

   a. Restore the inoperable channel(s) to operable status; or

   b. Place the inoperable channel(s) or associated trip system in the trip condition (not applicable if the inoperable channel is the result of an inoperable mechanical vacuum pump isolation valve).

F. Mechanical Vacuum Pump Isolation Instrumentation (4.2)

The High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be checked, functionally tested and calibrated as indicated in Surveillance Requirements 4.2.F.1, 2, 3, 4 and 5.

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required actions may be delayed for up to six (6) hours provided the associated trip function maintains mechanical vacuum pump isolation capability.

1. Perform an instrument check once each day.

2. Perform an instrument functional test once every three (3) months.

<MOVE TO SEPARATE PAGE>

A.5

WHEN A ROD BLOCK MONITOR CHANNEL IS PLACED IN AN INOPERABLE STATUS SOLELY FOR PERFORMANCE OF REQUIRED SURVEILLANCES, ENTRY INTO ASSOCIATED LIMITING CONDITIONS FOR OPERATION AND REQUIRED ACTIONS MAY BE DELAYED FOR UP TO 6 HOURS PROVIDED THE ASSOCIATED TRIP FUNCTION MAINTAINS CONTROL ROD BLOCK INITIATION CAPABILITY.

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TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

	Required Channels		Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run A.3	Trip Setting
	A.6		Rod Block Monitor (RBM A/B) LA.1				
1.a	(Notes 9 and 10)	2	a. Upscale (Flow Bias) (Note 7) A.3			X	≤0.66(W-ΔW)+N with a maximum as defined in the COLR (Note 5) LA.3
1.b	A.5	2	b. Downscale (Note 7) LA.2			X	≥2/125 Full Scale
1.c		2	c. Inop (Note 7)			X	
2	(Note 12) A.6	2	Reactor Mode Switch - Shutdown Position (Note 12) A.3				
	~~(Note 8)~~	~~1~~	~~Trip System~~	~~X~~	~~X~~	~~X~~	A.4

A.1

ACTION

TABLE 3.2.5 NOTES

1. Deleted.

2. Deleted.

3. Deleted.

4. Deleted.

LA.3

5. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. Refer to the Core Operating Limits Report for acceptable values for N. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. ΔW = 0 for two recirculation loop operation.

TABLE 3.2.5 FUNCTION 1.a TRIP SETTING

6. Not used.

A.3

7. The trip may be bypassed when the reactor power is <30% of Rated Thermal Power. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

LA.2

8. With the number of operable channels less than the required number, place the inoperable channel in the tripped condition within one hour.

A.4

ACTION NOTE 1

9. With one or two RBM channels inoperable: TAKE ALL OF THE APPLICABLE ACTIONS IN NOTES 1.a AND 1.b BELOW.

a. Verify that the reactor is not operating on a limiting control rod pattern (as described in the Bases for Specification 3.3.B.6); and

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ACTION NOTE 1.a

b. If one RBM channel is inoperable, restore the inoperable channel to operable status within 24 hours; and

ACTION NOTE 1.b

c. If the required is action(s) and associated completion time(s) of Notes 1 9.a and 9.b above are not met, or if two RBM channels are inoperable, place one RBM channel in the tripped condition within the next hour.

10. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required action notes may be delayed for up to 6 hours provided the associated Trip Function maintains Control Rod Block initiation capability.

A.5

11. Deleted.

12. Required to be operable when the reactor mode switch is in the shutdown position.

A.3

ACTION NOTE 2

13. With one or more Reactor Mode Switch - Shutdown Position channels inoperable, immediately suspend control rod withdrawal and immediately initiate action(s) to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

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TABLE 4.2.5

~~MINIMUM~~ TESTS AND ~~CALIBRATION~~ FREQUENCIES

CONTROL ROD BLOCK INSTRUMENTATION

	Trip Function	Functional Test	Calibration
	Rod Block Monitor (RBM)		(2) L.2
1.a	a. Upscale (Flow Bias)	Every Three Months ~~(Note 4)~~	Every Three Months ~~(Note 13)~~ LA.4
1.b	b. Downscale	Every Three Months ~~(Note 4)~~ A.8	Every Three Months (2) L.2
1.c	c. Inop	Every Three Months	
	Trip System Logic	Once/Operating Cycle	Once/Operating Cycle ~~(Note 3)~~ A.7 A.4
2.	Reactor Mode Switch - Shutdown Position	Every Refueling Outage (Note 12)	

(2) NEUTRON DETECTORS ARE EXCLUDED L.2

A.1

TABLE 4.2 NOTES

1. ~~Not used.~~

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart. A.9

3. ~~Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.~~ A.7

4. ~~This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.~~ A.8

5. ~~Deleted.~~

6. ~~Deleted.~~

7. ~~Deleted.~~

8. Functional tests and calibrations are not required when systems are not required to be operable. A.9

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

TABLE 4.2.5 FOOTNOTE (1)

12. ~~Trip system logic testing is not applicable to this function.~~ A.4 If the required surveillance frequency (every Refueling Outage) is not met, functional testing of the Reactor Mode Switch-Shutdown Position function shall be ~~initiated~~ PERFORMED M.1 within 1 hour after the reactor mode switch is placed in Shutdown ~~for the purpose of commencing a scheduled Refueling Outage.~~ A.1

13. ~~Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).~~ LA.4

3.3 LIMITING CONDITIONS FOR OPERATION	4.3 SURVEILLANCE REQUIREMENTS
pressure, are fully inserted, no more than two rods may be moved.	
	(c) Out-of-sequence control rods in each distinct RWM group shall be selected and the annunciator of the selection errors verified.
	(d) An out-of-sequence control rod shall be withdrawn no more than three notches and the rod block function verified.
4. Control rod patterns and the sequence of withdrawal or insertion shall be established such that the rod drop accident limit of 280 cal/g is not exceeded.	4. The control rod pattern and sequence of withdrawal or insertion shall be verified to comply with Specification 3.3.B.4.
5. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate greater than or equal to three counts per second.	5. Prior to control rod withdrawal for startup or during refueling, verification shall be made that at least two source range channels have an observed count rate of at least three counts per second.
6. ~~During operation with limiting control rod patterns either:~~ ~~(a) Both RBM channels shall be operable; or~~ ~~(b) Control rod withdrawal shall be blocked; or~~ Deleted. L.1	6. ~~When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.~~ Deleted. L.1

3.3 LIMITING CONDITIONS FOR OPERATION

(c) ~~The operating power level shall be limited so that the MCPR will remain above the fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.~~

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C. Scram Insertion Times

1.1 The average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.358
36	25.34	0.912
26	46.18	1.468
06	87.84	2.686

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.379
36	25.34	0.967
26	46.18	1.556
06	87.84	2.848

4.3 SURVEILLANCE REQUIREMENTS

7. The scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing under administrative control.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power with reactor pressure above 800 psig all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. During or following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the start up test report.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system provides a scram function in noncoincident configuration. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the continuous withdrawal of a designated single control rod could result in a violation of the MCPR safety limit or the 1% plastic strain limit. ~~A limiting control rod pattern is a pattern which results in the core being on a thermal limit (i.e., operating on a limiting value for APLHGR, LHGR, or MCPR. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.~~

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3.11 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

1. During operation at $\geq$25% Rated Thermal Power the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. For core flows other than rated, the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is provided in the Core Operating Limits Report. If at any time during operation at $\geq$25% Rated Thermal Power it is determined by normal surveillance that the limiting value for MCPR is being exceeded, MCPR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor power shall be brought to <25% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined once within 12 hours after $\geq$25% Rated Thermal Power, daily during operation at $\geq$25% Rated Thermal Power thereafter~~, and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.6~~.

L.1

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily when operating at ≥25% Rated Thermal Power to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. The 12 hour allowance after thermal power ≥25% Rated Thermal Power is achieved is acceptable given the large inherent margin to operating limits at low power levels.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

If a reactor power distribution limit is exceeded, an assumption regarding an initial condition of the DBA analysis, transient analyses, or the fuel design analysis may not be met. Therefore, prompt action should be taken to restore the APLHGR, LHGR or MCPR to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour completion time is sufficient to restore the APLHGR, LHGR, or MCPR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR, LHGR, or MCPR out of specification.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. ~~The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.~~ L.1

Safety Assessment

Discussion of Changes

3.2.E and 4.2.E

Control Rod Block Actuation

SAFETY ASSESSMENT OF CHANGES
TS 3.2.E/4.2.E - CONTROL ROD BLOCK INSTRUMENTATION

ADMINISTRATIVE

A.1 In the revision of the Vermont Yankee Nuclear Power Station (VYNPS) current Technical Specifications (CTS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the VYNPS Technical Specifications (TS) more consistent with human factor principles used in the Boiling Water Reactor Improved Standard Technical Specifications (ISTS), NUREG-1433, Rev. 2. These format and presentation changes are being made to improve usability and clarity. The changes are considered administrative.

A.2 CTS 3.2.D, 4.2.D, Table 3.2.5, Table 4.2.5 and associated Notes provide requirements that apply to off-gas isolation instrumentation. Changes these CTS off-gas isolation instrumentation requirements are addressed in the Safety Assessments of Changes for CTS 3.2.D/4.2.D, Off-Gas Isolation Instrumentation. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.3 CTS 3.2.E indicates that, during power operation, the control rod block instrumentation shall be operable in accordance with Table 3.2.5. The Rod Block Monitor (RBM) Trip Functions (proposed Table 3.2.5, Trip Functions 1.a, 1.b and 1.c) are required by CTS Table 3.2.5 to be operable in the Run Mode. CTS Table 3.2.5 Note 7 (first sentence) applies to these Trip Functions and indicates that these Trip Functions may be bypassed when reactor power is $\leq$ 30% of Rated Thermal Power. The intent of this note is to waive the operability requirements of the RBM Trip Functions when reactor power is $\leq$ 30%. The "APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS" column in proposed Table 3.2.5 requires the RBM Trip Functions to be operable when reactor power is > 30% RATED THERMAL POWER which is equivalent to CTS requirements. For the Reactor Mode Switch – Shutdown Position Trip Function, CTS Table 3.2.5 Note 12 modifies the requirements and indicates that this Trip Function is required to be operable when the reactor mode switch is in the shutdown position. The "APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS" column in proposed Table 3.2.5 requires the Reactor Mode Switch – Shutdown Position to be operable as identified in proposed Table 3.2.5 Footnote 1. This Footnote states, "When the reactor mode switch is in the shutdown position. As such, these changes are considered administrative in nature.

A.4 CTS 4.2.E specifies that instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5. In proposed Surveillance Requirement (SR) 4.2.E.1, the reference to "and logic system," is deleted since associated logic systems are considered part of the control rod block instrumentation Trip Functions included in proposed Tables 3.2.5 and 4.2.5. In addition, proposed Table 3.2.5 will delete explicit reference to "Trip System Logic" as a separate Trip Function. The requirement that the "Trip System Logic" be operable is required by both the definition of Operable and the TS requirements for the control rod block instrumentation Trip Functions to be operable, without explicit reference to "Trip System Logic." The control rod block instrumentation design at VYNPS only includes one "Trip System Logic." Therefore, when the "Trip System

ADMINISTRATIVE

A.4 (continued) Logic" is inoperable, both RBM channels would be inoperable and proposed Table 3.2.5 Action Note 1.b, would require one channel to be trip in one hour, which is equivalent to the actions in CTS Table 3.2.5 Note 8 (which is also proposed to be deleted as part of this change). For the purpose of testing, it is not necessary to explicitly identify "Trip System Logic" in CTS Table 4.2.5 for this design, since proposed Table 4.2.5 continues to require performance of surveillance testing of the "Trip System Logic" through the requirements for performance of Instrument Functional Tests. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.5 CTS Table 3.2.5 Note 10 provides an allowance to delay entry into actions for 6 hours for the situation of a RBM channel inoperable solely for performance of surveillances. This allowance is moved to proposed SR 4.2.E.1. This change does not involve a technical change, but is only a difference of presentation preference. Therefore, this change is considered administrative.

A.6 Notes 9 and 13 to CTS Table 3.2.5 provide actions when the required channels requirement is not met for the associated Trip Functions. These requirements are identified in a separate column in proposed Table 3.2.5 titled, "ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE," for each of the associated Trip Functions This change is a presentation preference and does not alter the current action requirements when required control rod block instrumentation channels are inoperable. Therefore, this change is considered administrative in nature.

A.7 For the Trip System Logic associated with the RBM control rod block instrumentation, CTS Table 4.2.5 includes requirements to perform a calibration of Trip System Logics once per Operating Cycle. These requirements are modified by Table 4.2 Note 3. Note 3 states, "Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system." The control rod block instrumentation Trip Functions of CTS Table 4.2.5 do not include any time delay relays or timers necessary for proper functioning of the trip systems. Therefore, this Note is deleted and, in proposed Table 4.2.5, the control rod block instrumentation (proposed Trip Functions 1.a, .b, 1.c, and 2) do not include calibration requirements for time delay relays or timers. As a result, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.8 CTS Table 4.2.5 includes Functional Test requirements for the RBM Upscale and Downscale Trip Functions. These requirements are modified by CTS Table 4.2 Note 4. Note 4 states, "This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel." The definition of Instrument Functional Test for this type of instrumentation (CTS 1.0.G.1) is, "the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions." The requirements of CTS Table

ADMINISTRATIVE

A.8 (continued) 4.2 Note 4 are consistent with the requirements of the Instrument Functional Test definition. The CTS definition of Instrument Functional Test allows the method of testing described in CTS Table 4.2 Note 4 to be used. Therefore, CTS Table 4.2 Note 4 is unnecessary and is deleted. This change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

A.9 CTS Table 4.2 Notes 2, 8, 10, and 11 provide requirements that apply to ECCS instrumentation. The ECCS instrumentation is located in proposed Specifications 3.2.A and 4.2.A. Therefore, the requirements of CTS Table 4.2 Notes 2, 8, 10, and 11 are physically moved and addressed in the changes for proposed Specifications 3.2.A and 4.2.A. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS Table 4.2 Note 9 provides requirements that apply to post-accident monitoring instrumentation. The post-accident monitoring instrumentation is located in proposed Specifications 3.2.G and 4.2.G. Therefore, the requirements of CTS Table 4.2 Note 9 are physically moved and changes addressed in proposed Specifications 3.2.G and 4.2.G. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Table 4.2 Note 12 applies to the Reactor Mode Switch – Shutdown Position Trip Function. This note allows the Functional Test of the Reactor Mode Switch – Shutdown Position Trip Function to be initiated with 1 hour after the reactor mode switch in placed in shutdown. This note does include a time limit on completion of the Functional Test. Since testing of this Trip Function with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links, the intent of this allowance is to ensure the required Functional Test is completed in a timely manner as soon as plant conditions exist to perform the test. Therefore, in proposed Table 4.2.5 Footnote (1), the allowance is revised by requiring the Functional Test of this Trip Function to be performed (i.e., completed) within 1 hour after the reactor mode switch is placed in the shutdown position. The 1 hour allowance continues to provide a reasonable time in which to complete the required Functional Test. This change represents an additional restriction on plant operation necessary to ensure that the Functional Test is satisfactorily completed in a timely manner. The change is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 3.2.5 details relating to system design and operation (i.e., the specific instrument tag numbers) are unnecessary in the TS and are proposed to be relocated to the Technical Requirements Manual (TRM). Proposed Specification 3.2.E and Table 3.2.5 require the control rod block instrumentation Trip Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.5 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required control rod block instrumentation Trip Functions are maintained operable. As such, the relocated details are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TRM are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.2 The LPRM inputs for operability of the RBM are relocated to Specification 3.2.E Bases. The Specification 3.2.E Bases indicates that if sufficient LPRMs are not available (the same requirements as specified in CTS Table 3.2.5, Note 7, second sentence), then the associated RBM is inoperable. As such, CTS Table 3.2.5 Note 7, second sentence, is not necessary in VYNPS TS control rod block instrumentation Table 3.2.5. The definition of operability suffices. Therefore, the relocated details of the note are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TS Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.3 CTS Table 3.2.5 Note 5 contains design and operational details of the RBM Upscale (Flow Bias) Trip Function Trip Setting for two recirculation loop and single recirculation loop operation. These details are not necessary to ensure the operability of the RBM Upscale (Flow Bias) Trip Function. Therefore, the information in this note is to be relocated to Specification 3.2.E Bases and reference to this information is deleted from VYNPS TS. The requirements of Specification 3.2.E and associated Table 3.2.5 which includes RBM Upscale (Flow Bias) Trip Function Trip Settings for both two recirculation loop and single recirculation loop operation are adequate to ensure the RBM Upscale (Flow Bias) Trip Function is maintained operable. As such, these relocated requirements are not required to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the TS Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LA.4 CTS Table 4.2.5 and associated Note 13 describe details of the performance of Instrument Calibrations of the RBM Upscale (Flow Bias) Trip Function. These details are to be relocated to Bases. These details are not necessary to ensure the operability of the RBM Upscale (Flow Bias) Trip Function instrumentation. The VYNPS TS definition of Instrument Calibration, the requirements of proposed Specification 3.2.E, and the associated Surveillance Requirements (including the requirements to periodically perform Instrument Calibrations) are adequate to ensure the RBM Upscale (Flow Bias) Trip Function instrumentation is maintained operable. As such, these relocated details are not required

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.4 (continued) to be in the VYNPS TS to provide adequate protection of the public health and safety. Changes to the Bases are controlled by the provisions of 10 CFR 50.59.

"Specific"

L.1 CTS Table 3.2.5 Note 9.a, which requires verification that the reactor is not operating on a limiting control rod pattern when one or two RBM channels are inoperable, CTS 3.3.B.6, which provides additional RBM requirements when operating on a limiting control rod pattern, CTS 4.3.B.6, which requires a Functional Test of the RBM prior to control rod withdrawal and daily thereafter when the reactor is operating on a limiting control rod pattern and the CTS 4.11.C requirement to determine Minimum Critical Power Ratio (MCPR) following any change in power level or distribution that would cause operation on a limiting control rod pattern, have been deleted. Corresponding changes are also made to the associated Bases. Since a limiting control rod pattern is operation on a power distribution limit (such as Average Planar Linear Heat Generation Rate or MCPR), the condition is extremely unlikely. It is also highly unlikely that, when operating on a power distribution limit, the operator would withdraw a control rod. The status of power distribution limits does not affect the operability of the RBM and therefore, no additional requirements on the RBM System are necessary (e.g., that it be tripped within one hour with a channel inoperable while on a limiting control rod pattern or that both RBM channels be operable while on a limiting control rod pattern). Adequate requirements on power distribution limits are specified in the Specifications in TS 3.11/4.11, Reactor Fuel Assemblies, without the need for additional determination of MCPR when operating on a limiting control rod pattern. Furthermore, due to the improbability of operating exactly on a thermal limit, the CTS Actions and Surveillance Requirements would almost never be required. In addition, since the Surveillance Requirements in CTS 4.3.B.6 and CTS 4.11.C are not specific as to when "prior to" or "following" operation on a limiting control rod pattern, respectively, they are required to be performed, these two frequency requirements could be satisfied by the normal periodic Surveillances. The additional requirements associated with operation on a limiting control rod pattern are not necessary to ensure that the RBM channels are maintained operable or that power distribution parameters are maintained within limits. Therefore, the deletion of the CTS requirements associated with operation on a limiting control rod pattern is not safety significant. This change is consistent with the ISTS.

L.2 This change adds a note to the RBM Instrument Calibrations in CTS Table 4.2.5 excluding the neutron detectors from this Surveillance (proposed Table 4.2.5 Footnote (2)). The Instrument Calibration is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. The neutron detectors are excluded from the RBM Instrument Calibration because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. In addition, changes in neutron detector sensitivity are compensated for by performance of the 7 day heat balance calibration and the 2000 MWD/T Local Power Range Monitor calibration using the Traversing Incore Probe System of the VYNPS Reactor Protection System TS. This change is consistent with the ISTS.

RELOCATED SPECIFICATIONS

None

No Significant Hazards Consideration

3.2.E and 4.2.E

Control Rod Block Actuation

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, PROCEDURES, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents. The Bases, UFSAR, procedures, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the relocated details in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant reduction in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specifications, NUREG-1433, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.E/4.2.E - CONTROL ROD BLOCK INSTRUMENTATION

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the additional Rod Block Monitor (RBM) and Minimum Critical Power Ratio (MCPR) action and surveillance requirements associated with operating on a limiting control rod pattern. The RBMs and MCPR are not assumed to be initiators of any analyzed event. During operation on a limiting control rod pattern, power distribution parameters continue to meet acceptable required limits. In addition, the status of power distribution limits (i.e., operation on a limiting control rod pattern) does not affect the operability or reliability of the RBMs. Technical Specifications continue to provide assurance that power distribution parameters are maintained within limits and the RBMs are maintained capable of performing their required function. As a result, the consequences of an accident occurring with the proposed change are the same as the consequences occurring with the current requirements. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change continues to ensure the affected instrumentation is capable of performing its function as assumed in the safety analyses and that power distribution parameters are maintained within required limits. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   Deleting the additional Rod Block Monitor (RBM) and Minimum Critical Power Ratio (MCPR) action and surveillance requirements associated with operating on a limiting control rod pattern does not involve a significant reduction in a margin of safety. This change is acceptable because it does not impact the ability of the RBMs to perform their intended safety function or the ability to maintain power distribution parameters within required limits. Since Technical Specifications continue to provide adequate assurance of RBM operability and maintenance of power distribution parameters within required limits, the ability to satisfy safety analysis assumptions and initial conditions is unchanged. In addition, the probability of actually operating on a limiting control rod pattern is very low. Since the change has no effect on any safety analysis assumptions or initial conditions, the margins of safety continue to be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change excludes neutron detectors from the Rod Block Monitor (RBM) Instrument Calibration Surveillance Requirements. The RBM Instrumentation and associated Surveillance Requirements are not assumed to be initiators of any analyzed event. The Technical Specifications will continue to ensure that changes in the associated neutron detectors are compensated for in other periodic Surveillance Requirements. As such, the proposed change will not affect the ability of the RBM to perform its safety function of preventing a continuous rod withdrawal error event and accident consequences are not changed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change continues to ensure the affected instrumentation is capable of performing its function as assumed in the safety analyses. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   Excluding the neutron detectors from the RBM Instrument Calibration does not involve a significant reduction in a margin of safety. The Technical Specifications will continue to ensure the RBM is capable of performing its safety function. Changes in the neutron detectors, excluded from the RBM Instrument Calibration by this proposed change, are compensated for in other periodic Technical Specification Surveillance Requirements. As such, the proposed change will not affect the ability of the RBM to perform its safety function of preventing a continuous rod withdrawal error event. Since the change has no effect on any safety analysis assumptions or initial conditions, the margins of safety continue to be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

References

3.2.E and 4.2.E

Control Rod Block Actuation

3.2.E/4.2.E REFERENCES
Control Rod Block Actuation

1. UFSAR, Section 7.5.8.

2. UFSAR, Section 7.7.4.3.2.

3. UFSAR, Section 14.5.3.1.

4. GENE-770-06-1-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

5. NEDC-30851P-A Supplement 1, "Technical Specifications Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

Proposed

Technical Specifications

3.2.F and 4.2.F

Mechanical Vacuum Pump Isolation Instrumentation

3.2 LIMITING CONDITIONS FOR OPERATION

F. Mechanical Vacuum Pump Isolation Instrumentation

1. When the reactor is in the RUN or STARTUP/HOT STANDBY Mode and the mechanical vacuum pump is in service, 4 channels of the High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be operable.

4.2 SURVEILLANCE REQUIREMENTS

F. Mechanical Vacuum Pump Isolation Instrumentation

1. The High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be checked, functionally tested and calibrated as indicated in Surveillance Requirements 4.2.F.1.a, b, c, d and e.

   When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains mechanical vacuum pump isolation capability.

   a. Perform an Instrument Check once each day.

   b. Perform an Instrument Functional Test once every 3 months.

   c. Perform an Instrument Calibration, except for radiation detectors, using a current source once every 3 months. The Trip Setting shall be $\leq$ 3.0 X background at rated thermal power.

   d. Perform an Instrument Calibration using a radiation source once each Refueling Outage.

   e. Perform a Logic System Functional Test, including mechanical vacuum pump isolation valve, once every Operating Cycle.

3.2 LIMITING CONDITIONS FOR OPERATION

2. If Specification 3.2.F.1 is not met, take all of the applicable Actions in Specifications 3.2.F.2.a and 2.b below.

   a. With one or more channels inoperable:

      1) Restore the inoperable channel(s) to operable status within 12 hours; or

      2) Place the inoperable channel(s) or associated trip system in the trip condition within 12 hours (not applicable if the inoperable channel is the result of an inoperable mechanical vacuum pump isolation valve).

   b. If the required Action and associated completion time of Specification 3.2.F.2.a above is not met, or if mechanical vacuum pump isolation capability is not maintained:

      1) Isolate the mechanical vacuum pump within 12 hours; or

      2) Isolate the main steam lines within 12 hours; or

      3) Place the reactor in the SHUTDOWN Mode within 12 hours.

4.2 SURVEILLANCE REQUIREMENTS

Proposed

Bases

3.2.F and 4.2.F

Mechanical Vacuum Pump Isolation Instrumentation

BASES: 3.2.F/4.2.F MECHANICAL VACUUM PUMP ISOLATION

BACKGROUND

The mechanical vacuum pump isolation instrumentation initiates an isolation of the mechanical vacuum pump following events in which main steam radiation monitors exceed a predetermined value. Tripping and isolating the mechanical vacuum pumps limits control room and offsite doses in the event of a control rod drop accident (CRDA).

The mechanical vacuum pump isolation instrumentation includes sensors, relays and switches that are necessary to cause initiation of mechanical vacuum pump isolation. The channels include electronic equipment that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an isolation signal to the mechanical vacuum pump isolation logic.

The isolation logic consists of two independent trip systems, with two channels of the High Main Steam Line Radiation Trip Function in each trip system. Each trip system is a one-out-of-two logic for this Trip Function. Thus, either channel of the High Main Steam Line Radiation Trip Function in a trip system is needed to trip the trip system. The outputs of the channels in a trip system are arranged in a logic so that both trip systems must trip to result in an isolation signal.

The mechanical vacuum pump isolation valve is also associated with this Trip Function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The mechanical vacuum pump isolation is assumed in the safety analysis for the CRDA. The mechanical vacuum pump isolation instrumentation initiates an isolation of the mechanical vacuum pump to limit control room and offsite doses resulting from fuel cladding failure in a CRDA.

The mechanical vacuum pump isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The operability of the mechanical vacuum pump isolation instrumentation is dependent on the operability of the four High Main Steam Line Radiation Trip Function instrumentation channels with their trip setpoints within the calculational as-found tolerances specified in plant procedures. Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Surveillance Requirement 4.2.F.1.c as required by the CRDA analysis. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions. The High Main Steam Line Radiation Trip Setting was chosen to be as low enough to ensure that control room and offsite dose limits are not exceeded in the event of a CRDA, but high enough to avoid spurious isolation due to nitrogen-16 spikes, instrument instabilities, and other operational occurrences. Channel operability also includes the mechanical vacuum pump isolation valve.

BASES: 3.2.F/4.2.F MECHANICAL VACUUM PUMP ISOLATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The mechanical vacuum pump isolation is required to be operable in RUN and STARTUP/HOT STANDBY when the mechanical vacuum pump is in service to mitigate the consequences of a postulated CRDA. In this condition, fission products released during a CRDA could be discharged directly to the environment. Therefore, the mechanical vacuum pump isolation is necessary to assure conformance with the radiological evaluation of the CRDA. In other Modes or conditions, the consequences of a control rod drop are insignificant, and are not expected to result in any fuel damage or fission product releases. When the mechanical vacuum pump is not in operation in RUN and STARTUP/HOT STANDBY, fission product releases via this pathway would not occur.

ACTIONS

Specification 3.2.F.2.a

With one or more High Main Steam Line Radiation Trip Function channels inoperable, but with mechanical vacuum pump isolation capability maintained (refer to Specification 3.2.F.2.b Bases), the mechanical vacuum pump isolation instrumentation is capable of performing the intended function. However, the reliability and redundancy of the mechanical vacuum pump isolation instrumentation is reduced, such that a single failure in one of the remaining channels could result in the inability of the mechanical vacuum pump isolation instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to operable status. Because of the low probability of an extensive number of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of mechanical vacuum pump isolation, 12 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to operable status. Alternately, the inoperable channel or associated trip system may be placed in trip, since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable mechanical vacuum pump isolation valve, since this may not adequately compensate for the inoperable mechanical vacuum pump isolation valve (e.g., the isolation valve may be inoperable such that it will not close). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel would result in loss of condenser vacuum), or if the inoperable channel is the result of an inoperable mechanical vacuum pump isolation valve, Specification 3.2.F.2.b must be entered and its required actions taken.

Specification 3.2.F.2.b

With any required Action and associated completion time of Specification 3.2.F.2.a not met, the plant must be brought to a Mode or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least SHUTDOWN within 12 hours. Alternately, the mechanical vacuum pump may be isolated since this performs the intended function of the instrumentation. An additional option is provided to isolate

BASES: 3.2.F/4.2.F MECHANICAL VACUUM PUMP ISOLATION

ACTIONS (continued)

the main steam lines, which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser. This isolation is accomplished by isolation of all main steam lines and main steam line drains which bypass the main steam isolation valves.

Specification 3.2.F.2.b is also intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels result in the High Main Steam Line Radiation Trip Function not maintaining mechanical vacuum pump isolation capability. The High Main Steam Line Radiation Trip Function is considered to be maintaining mechanical vacuum pump isolation capability when sufficient channels are operable or in trip such that the mechanical vacuum pump isolation instruments will generate a trip signal from a valid High Main Steam Line Radiation signal, and the mechanical vacuum pump will be isolated. This requires one channel of the High Main Steam Line Radiation Trip Function in each trip system to be operable or in trip, and the mechanical vacuum pump isolation valve to be operable.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.F.1

As indicated in Surveillance Requirement 4.2.F.1, the High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be checked, functionally tested and calibrated as indicated Surveillance Requirements 4.2.F.1.a, b, c, d, and e.

Surveillance Requirement 4.2.F.1 also indicates that when a channel is placed in an inoperable status solely for performance of required instrumentation Surveillances, entry into associated LCO and required Actions may be delayed for up to 6 hours provided the associated Trip Function maintains mechanical vacuum pump isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable LCO entered and required Actions taken. This allowance is based on the reliability analysis (Ref. 1) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a mechanical vacuum pump will isolate when necessary.

Surveillance Requirement 4.2.F.1.a, Instrument Check

Performance of an Instrument Check once each day ensures that a gross failure of instrumentation has not occurred. An Instrument Check is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. An Instrument Check will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each Calibration. Agreement criteria are determined by the plant staff based

BASES: 3.2.F/4.2.F MECHANICAL VACUUM PUMP ISOLATION

SURVEILLANCE REQUIREMENTS (continued)

on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit. The Frequency is based upon operating experience that demonstrates channel failure is rare. The Instrument Check supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

Surveillance Requirement 4.2.F.1.b, Instrument Functional Test

An Instrument Functional Test is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of "once every 3 months" is based on the reliability analysis of Reference 1.

Surveillance Requirements 4.2.F.1.c and 4.2.F.1.d, Instrument Calibrations

An Instrument Calibration is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. An Instrument Calibration leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. Surveillance Requirement 4.2.F.1.c requires a calibration to be performed once every 3 months using a current source. This current source is provided downstream of the radiation detectors. As such, the radiation detectors are excluded from the 3 month calibration. Surveillance Requirement 4.2.F.1.d requires a calibration to be performed once each Refueling Outage using a radiation source. The radiation detectors are included in the once each Refueling Outage calibration. The specified Instrument Calibration Frequencies are based upon the time interval assumptions for calibration used in the determination of the magnitude of equipment drift in the associated setpoint analyses.

Surveillance Requirement 4.2.F.1.e, Logic System Functional Test

The Logic System Functional Test demonstrates the operability of the required trip logic for a specific channel. Actuation of the mechanical vacuum pump isolation valve is included as part of this Surveillance to provide complete testing of the assumed safety function. Therefore, if the isolation valve is incapable of actuating, the instrument channel would be inoperable. The Frequency of "once every Operating Cycle" is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the specified Frequency.

BASES: 3.2.F/4.2.F MECHANICAL VACUUM PUMP ISOLATION

REFERENCES

1. NEDC-30851P-A, Supplement 2, Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation, March 1989.

Current

Technical Specifications

Markups

3.2.F and 4.2.F

Mechanical Vacuum Pump Isolation Instrumentation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

D. Off-Gas System Isolation

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

E. Control Rod Block Actuation

During reactor power operation the instrumentation that initiates control rod block shall be operable in accordance with Table 3.2.5.

F. Mechanical Vacuum Pump Isolation Instrumentation

1. When the reactor is in the RUN or STARTUP/HOT STANDBY Mode and the mechanical vacuum pump is in service, ~~four~~ (4) channels of the High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be operable. ~~except as provided below.~~

a ~~1~~. With one or more channels inoperable ~~within 12 hours:~~

1) ~~a~~ Restore the inoperable channel(s) to operable status~~; or~~ WITHIN 12 HOURS

2) ~~b~~ Place the inoperable channel(s) or associated trip system in the trip condition (not applicable if the inoperable channel is the result of an inoperable mechanical vacuum pump isolation valve).

2. IF SPECIFICATION 3.2.F.1 IS NOT MET, TAKE ALL OF THE APPLICABLE ACTIONS IN SPECIFICATIONS 3.2.F.2.a AND 2.b BELOW.

4.2 SURVEILLANCE REQUIREMENTS

D. Off-Gas System Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.

A.2

<MOVE TO SEPARATE PAGE>

E. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5.

F. Mechanical Vacuum Pump Isolation Instrumentation

1. The High Main Steam Line Radiation Trip Function for mechanical vacuum pump isolation shall be checked, functionally tested and calibrated as indicated in Surveillance Requirements 4.2.F.1, ~~2~~, ~~3~~, ~~4~~ and ~~5~~. a b c d e

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ~~A~~actions may be delayed for up to ~~six~~ (6) hours provided the associated ~~T~~trip ~~F~~function maintains mechanical vacuum pump isolation capability.

a ~~2~~. Perform an ~~I~~instrument ~~C~~check once each day.

b ~~3~~. Perform an ~~I~~instrument ~~F~~functional ~~T~~test once every ~~three~~ (3) months.

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

b. ~~7~~ 1. If the required ~~A~~action and associated completion time of Specification 3.2.F.~~7~~ 2.a ABOVE is not met, ~~within the following 12 hours:~~

OR IF MECHANICAL VACUUM PUMP ISOLATION CAPABILITY IS NOT MAINTAINED:

M.1

1) ~~a~~ Isolate the mechanical vacuum pump WITHIN 12 HOURS; or

2) ~~b~~ Isolate the main steam lines; or

3) ~~c~~ Place the reactor in the SHUTDOWN Mode.

4.2 SURVEILLANCE REQUIREMENTS

c. ~~2~~. Perform an ~~I~~instrument ~~C~~calibration, except for the radiation detectors, using a current source once every ~~three~~ 30 months. The ~~T~~trip ~~S~~setting shall be ≤ 3.0 ~~times~~ X background at rated thermal power.

d. ~~3~~. Perform an ~~I~~instrument ~~C~~calibration using a radiation source once each ~~R~~refueling ~~O~~outage.

e. ~~4~~. Perform a ~~L~~logic ~~S~~system ~~F~~functional ~~T~~test, including mechanical vacuum pump isolation valve, once ~~each~~ EVERY ~~O~~operating ~~C~~cycle.

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G. Post-Accident Instrumentation

During reactor power operation, the instrumentation that displays information in the Control Room necessary for the operator to initiate and control the systems used during and following a postulated accident or abnormal operating condition shall be operable in accordance with Table 3.2.6.

G. Post-Accident Instrumentation

The post-accident instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.6.

H. Drywell to Torus ΔP Instrumentation

1. During reactor power operation, the Drywell to Torus ΔP Instrumentation (recorder #1-156-3 and instrument DPI-1-158-6) shall be operable except as specified in 3.2.H.2.

2. From and after the date that one of the Drywell to Torus ΔP instruments is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless the instrument is

H. Drywell to Torus ΔP Instrumentation

The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

A.2

Safety Assessment

Discussion of Changes

3.2.F and 4.2.F

Mechanical Vacuum Pump Isolation Instrumentation

SAFETY ASSESSMENT OF CHANGES
TS 3.2.F/4.2.F – MECHANICAL VACUUM PUMP ISOLATION INSTRUMENTATION

ADMINISTRATIVE

A.1 In the revision of the Vermont Yankee Nuclear Power Station (VYNPS) current Technical Specifications (CTS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the VYNPS Technical Specifications (TS) more consistent with human factor principles used in the Boiling Water Reactor Improved Standard Technical Specifications (ISTS), NUREG-1433, Rev. 2. These format and presentation changes are being made to improve usability and clarity. The changes are considered administrative.

A.2 CTS 3.2.D and 4.2.D provide requirements that apply to off-gas isolation instrumentation. Changes these CTS off-gas isolation instrumentation requirements are addressed in the Safety Assessments of Changes for CTS 3.2.D/4.2.D, Off-Gas Isolation Instrumentation. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative. CTS 3.2.H and 4.2.H provide requirements that apply to drywell to torus ΔP instrumentation. Changes these CTS drywell to torus ΔP instrumentation requirements are addressed in the Safety Assessments of Changes for CTS 3.2.H/4.2.H, Drywell to Torus ΔP Instrumentation. Therefore, this change does not involve a technical change, but is only a difference of presentation preference and is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.2.F.1 provides actions for one or more mechanical vacuum pump isolation instrumentation channels inoperable. These actions provide 12 hours to restore the inoperable channel(s) or place them or their associated trip system in the trip condition. Proposed TS 3.2.F.2 is revised to address the condition of multiple, inoperable, untripped mechanical vacuum pump isolation instrumentation channels that result in the High Main Steam Line Radiation Trip Function not maintaining mechanical vacuum pump isolation capability. The proposed change will eliminate the 12 hour allowed outage time for this condition. In this condition (i.e., loss of mechanical vacuum pump isolation capability), continued operation in accordance with the 12 hour allowed outage time is not appropriate, nor consistent with the applicable analyses of NEDC-30851P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation." This change is an additional restriction on plant operation necessary to achieve consistency with applicable analyses.

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

None

No Significant Hazards Consideration

3.2.F and 4.2.F

Mechanical Vacuum Pump Isolation Instrumentation

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee Nuclear Power Station has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.F/4.2.F – MECHANICAL VACUUM PUMP ISOLATION INSTRUMENTATION

TECHNICAL CHANGES – LESS RESTRICTIVE

There were no specific less restrictive changes identified for this Specification.

References

3.2.F and 4.2.F

Mechanical Vacuum Pump Isolation Instrumentation

3.2.F/4.2.F REFERENCE
Mechanical Vacuum Pump Isolation Instrumentation

1. NEDC-30851P-A, Supplement 2, Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation, March 1989.