Proposed

Technical Specifications

3.2.G and 4.2.G

Post-Accident Monitoring Instrumentation

3.2 LIMITING CONDITIONS FOR OPERATION

G. Post-Accident Monitoring Instrumentation

The post-accident monitoring instrumentation for each Function in Table 3.2.6 shall be operable in accordance with Table 3.2.6.

4.2 SURVEILLANCE REQUIREMENTS

G. Post-Accident Monitoring Instrumentation

1. The post-accident monitoring instrumentation shall be checked and calibrated in accordance with Table 4.2.6.

   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.

Table 3.2.6 (page 1 of 1)
Post-Accident Monitoring Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE
1. Drywell Atmospheric Temperature	RUN, STARTUP/HOT STANDBY	2	Note 1
2. Drywell Pressure	RUN, STARTUP/HOT STANDBY	2	Note 1
3. Torus Pressure	RUN, STARTUP/HOT STANDBY	2	Note 1
4. Torus Water Level	RUN, STARTUP/HOT STANDBY	2	Note 1
5. Torus Water Temperature	RUN, STARTUP/HOT STANDBY	2	Note 1
6. Reactor Pressure	RUN, STARTUP/HOT STANDBY	2	Note 1
7. Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY	2	Note 1
8. Torus Air Temperature	RUN, STARTUP/HOT STANDBY	2	Note 1
9. Containment Hydrogen/Oxygen Monitor	RUN, STARTUP/HOT STANDBY	2	Note 1
10. Containment High Range Radiation Monitor	RUN, STARTUP/HOT STANDBY	2	Note 2

Table 3.2.6 ACTION Notes

1. With one or more Post-Accident Monitoring instrumentation channels, for Functions other than Function 10, inoperable, take all of the applicable Actions in Notes 1.a and 1.b below.

   a. With one or more Functions with one channel inoperable:

      1) Restore channel to operable status within 30 days; or
      2) Prepare and submit a special report to the Commission within the next 14 days, outlining the Action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to operable status.

   b. With one or more Functions with two channels inoperable:

      1) Restore one required channel to operable status within 7 days; or
      2) Place the reactor in HOT SHUTDOWN within the next 12 hours.

2. With one or more Post - Accident Monitoring instrumentation Function 10 channels inoperable, take all of the applicable Actions in Notes 2.a and 2.b below.

   a. With one channel inoperable:

      1) Restore channel to operable status within 30 days; or
      2) Prepare and submit a special report to the Commission within the next 14 days, outlining the Action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to operable status.

   b. With two channels inoperable:

      1) Restore one channel to operable status within 7 days; or
      2) Prepare and submit a special report to the Commission within the next 14 days, outlining the Action taken, the cause of the inoperability, and the plans and schedule for restoring the channels to operable status.

Table 4.2.6 (page 1 of 1)
Post-Accident Monitoring Instrumentation
Tests and Frequencies

FUNCTION	CHECK	CALIBRATION
1. Drywell Atmospheric Temperature	Once/Day	Every 6 Months
2. Drywell Pressure	Once/Day	Once/Operating Cycle
3. Torus Pressure	Once/Day	Once/Operating Cycle
4. Torus Water Level	Once/Day	Once/Operating Cycle
5. Torus Water Temperature	Once/Day	Every 6 Months
6. Reactor Pressure	Once/Day	Once/Operating Cycle
7. Reactor Vessel Water Level	Once/Day	Once/Operating Cycle
8. Torus Air Temperature	Once/Day	Every 6 Months
9. Containment Hydrogen/Oxygen Monitor	Once/Day	Once/Operating Cycle
10. Containment High Range Radiation Monitor	Once/Day	Once/Operating Cycle

Proposed

Bases

3.2.G and 4.2.G

Post-Accident Monitoring Instrumentation

BASES: 3.2.G/4.2.G POST-ACCIDENT MONITORING INSTRUMENTATION

BACKGROUND

The primary purpose of the post-accident monitoring (PAM) instrumentation is to display, in the control room, plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events. The instruments that monitor these variables are designated as Type A, Category I, and non-Type A, Category I, in accordance with Regulatory Guide 1.97 (Ref. 1).

The operability of the post-accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation Specification ensures the operability of Regulatory Guide 1.97, Type A variables so that the control room operating staff can:

- Perform the diagnosis specified in the Emergency Operating Procedures (EOPs). These variables are restricted to preplanned actions for the primary success path of Design Basis Accidents (DBAs), (e.g., loss of coolant accident (LOCA)), and

- Take the specified, preplanned, manually controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function.

The PAM instrumentation Specification also ensures operability of most Category I, non-Type A, variables so that the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;

- Determine the potential for causing a gross breach of the barriers to radioactivity release;

- Determine whether a gross breach of a barrier has occurred; and

- Initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

The plant specific Regulatory Guide 1.97 analysis (Ref. 2) documents the process that identified Type A and Category I, non-Type A, variables.

Post-accident monitoring instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2)(ii). Category I, non-Type A, instrumentation is retained in Technical Specifications (TS) because they are intended to assist operators in

BASES: 3.2.G/4.2.G POST-ACCIDENT MONITORING INSTRUMENTATION

APPLICABLE SAFETY ANALYSES (continued)

minimizing the consequences of accidents. Therefore, these Category I variables are important for reducing public risk.

LCO

Specification 3.2.G and Table 3.2.6 require two operable channels for each Function to ensure that no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following an accident. Furthermore, providing two channels allows an Instrument Check during the post accident phase to confirm the validity of displayed information.

The following list is a discussion of the specified instrument Functions listed in Table 3.2.6.

1. Drywell Atmospheric Temperature

Drywell atmospheric temperature is a Type A and Category I variable provided to detect a reactor coolant pressure boundary (RCPB) breach and to verify the effectiveness of Emergency Core Cooling System (ECCS) functions that operate to maintain containment integrity. Two redundant temperature signals are transmitted from separate temperature elements for each channel. The output of one of these channels is recorded on a recorder in a control room. The output of the other channel is displayed on an indicator in the control room. The drywell atmospheric temperature channels measure from 0°F to 350°F. Therefore, the PAM Specification deals specifically with this portion of the instrument channels.

2. Drywell Pressure

Drywell pressure is a Type A and Category I variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain Reactor Coolant System (RCS) integrity. Two drywell pressure signals are transmitted from separate pressure transmitters for each channel. The output of these channels is displayed on two independent indicators in the control room. The pressure channels measure from -15 psig to 260 psig. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

3. Torus Pressure

Torus pressure is a Type A and Category I variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two torus pressure signals are transmitted from separate pressure transmitters and displayed on two independent indicators in the control room. The range of

LCO (continued)

indication is - 15 psig to 65 psig. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

4. Torus Water Level

Torus water level is a Type A and Category I variable provided to detect a breach in the RCPB. This variable is also used to verify and provide long term surveillance of ECCS function. The Torus Water Level Function provides the operator with sufficient information to assess the status of both the RCPB and the water supply to the ECCS. The Torus Water Level Function channels monitor the torus water level from the bottom of the torus to 5 feet above the normal torus water level. Two torus water level signals are transmitted from separate level transmitters to two independent control room indicators in the control room. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

5. Torus Water Temperature

Torus water temperature is a Type A and Category I variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two redundant temperature signals are transmitted from separate temperature elements for each channel. The temperature channels output to two independent control room indicators. The range of the torus water temperature channels is 0°F to 250°F. Therefore, the PAM Specification deals specifically with this portion of the instrument channels.

6. Reactor Pressure

Reactor pressure is a Type A and Category I variable provided to support monitoring of RCS integrity and to verify operation of the ECCS. Two independent pressure transmitters, with a range of 0 psig to 1500 psig, monitor pressure and provide pressure indication to the control room. The output from these channels is provided to two independent indicators in the control room. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

7. Reactor Vessel Water Level

Reactor vessel water level is a Type A and Category I variable provided to support monitoring of core cooling and to verify operation of the ECCS. Water level is measured by independent differential pressure transmitters for each channel. Each channel measures from -200 inches to + 200 inches, referenced to the top of enriched fuel. The output from these channels is provided to two independent indicators in the control room. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

BASES: 3.2.G/4.2.G POST-ACCIDENT MONITORING INSTRUMENTATION

LCO (continued)

8. Torus Air Temperature

Torus air temperature is a Type A and Category I variable provided to detect a RCPB breach and to verify the effectiveness of ECCS functions that operate to maintain containment integrity. Two redundant temperature signals are transmitted from separate temperature elements for each channel. The output of one of these channels is recorded on a recorder in a control room with a range of 50°F to 300°F. The output of the other channel is displayed on an indicator in the control room with a range of 0°F to 350°F. Therefore, the PAM Specification deals specifically with this portion of the instrument channels.

9. Containment Hydrogen/Oxygen Concentration Monitor

Containment hydrogen and oxygen monitors are Category I instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. Hydrogen and oxygen concentrations are each measured by two redundant monitors whose output is provided to the control room. The containment hydrogen and oxygen monitor PAM instrumentation consists of two independent gas analyzers. Each gas analyzer consists of a hydrogen analyzer and an oxygen analyzer. The analyzers are capable of determining hydrogen concentration in the range of 0% to 30% and oxygen concentration in the range of 0% to 25%. There are two independent recorders in the control room to display the results.

10. Containment High Range Radiation Monitor

Containment high range radiation is a Category 1 variable provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Two redundant radiation detectors are mounted in the drywell. Each radiation detector provides a signal to an independent monitor in the control room, which has a range from 10^0 R/hr to 10^7 R/hr. The outputs of these radiation monitors are displayed on two independent indicators located in the control room. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

APPLICABILITY

The PAM instrumentation Specification is applicable in the RUN and STARTUP/HOT STANDBY Modes. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in the RUN and STARTUP/HOT STANDBY Modes. In other Modes and conditions, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be operable in these other Modes or conditions.

BASES: 3.2.G/4.2.G POST-ACCIDENT MONITORING INSTRUMENTATION

ACTIONS

Table 3.2.6 ACTION Note 1

Table 3.2.6 ACTION Note 1.a.1) requires that, when one or more Functions (except Function 10) have one required channel that is inoperable, the required inoperable channel must be restored to operable status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining operable channels, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval. If the inoperable channel of each affected Function has not been restored to operable status in 30 days, Table 3.2.6 ACTION Note 1.a.2) requires a special written report be submitted to the NRC within the next 14 days. The report will outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation to operable status. This action is appropriate in lieu of a shutdown requirement, since another operable channel is monitoring the Function, an alternate method of monitoring is available, and given the likelihood of plant conditions that would require information provided by this instrumentation.

Table 3.2.6 ACTION Note 1.b.1) requires that, when one or more Functions, except Function 10, have two required channels that are inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to operable status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. If at least one channel of each affected Function has not been restored to operable status in 7 days, Table 3.2.6 ACTION Note 1.b.2) requires the plant to be brought to a Mode in which the LCO does not apply. To achieve this status, the plant must be brought to at least HOT SHUTDOWN within 12 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.

Table 3.2.6 ACTION Note 2

Table 3.2.6 ACTION Note 2.a.1) requires that, when Function 10 has one required channel that is inoperable, the required inoperable channel must be restored to operable status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining operable channels, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval. If the inoperable channel has not been restored to operable status in 30 days, Table 3.2.6 ACTION Note 2.a.2) requires a special written report be submitted to the NRC within the next 14 days. The report will outline the preplanned alternate

BASES: 3.2.G/4.2.G POST-ACCIDENT MONITORING INSTRUMENTATION

ACTIONS (continued)

method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation to operable status. This action is appropriate in lieu of a shutdown requirement, since another operable channel is monitoring the Function, an alternate method of monitoring is available, and given the likelihood of plant conditions that would require information provided by this instrumentation.

Table 3.2.6 ACTION Note 2.b.1) requires that, when Function 10 has two required channels that are inoperable, one channel should be restored to operable status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Since alternate means of monitoring drywell radiation have been developed and tested, the action required by Table 3.2.6 ACTION 2.b.2), if at least one channel has not been restored to operable status within 7 days, is not to shut down the plant, but rather to submit a special written report to the NRC within the next 14 days. The report will outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the normal PAM instrumentation to operable status. The alternate means of monitoring may be temporarily installed if the normal PAM channel cannot be restored to operable status within the allotted time. The report provided to the NRC should also describe the degree to which the alternate means are equivalent to the installed PAM channels and justify the areas in which they are not equivalent.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.G.1

As indicated in Surveillance Requirement 4.2.G.1, post-accident monitoring instrumentation shall be checked and calibrated as indicated in Table 4.2.6. Table 4.2.6 identifies, for each Function, the applicable Surveillance Requirements.

Surveillance Requirement 4.2.G.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. 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. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring post-accident parameters, when necessary.

BASES: 3.2.G/4.2.G POST-ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (continued)

Table 4.2.6, 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 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 4.2.6, Calibration

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. The specified Instrument Calibration Frequencies are based on operating experience.

REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.

2. NRC letter, M.B. Fairtile (NRC) to L.A. Temblay (VYNPC), "Conformance to Regulatory Guide 1.97 for Vermont Yankee Nuclear Power Station," December 4, 1990.

Current

Technical Specifications

Markups

3.2.G and 4.2.G

Post-Accident Monitoring Instrumentation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION	4.2 SURVEILLANCE REQUIREMENTS
2. If the required action and associated completion time of Specification 3.2.F.1 is not met, within the following 12 hours: a. Isolate the mechanical vacuum pump; or b. Isolate the main steam lines; or c. Place the reactor in the SHUTDOWN Mode.	3. Perform an instrument calibration, except for the radiation detectors, using a current source once every three (3) months. The trip setting shall be ≤ 3.0 times background at rated thermal power. 4. Perform an instrument calibration using a radiation source once each refueling outage. 5. Perform a logic system functional test, including mechanical vacuum pump isolation valve, once each operating cycle.

<MOVE TO SEPARATE PAGE>

MONITORING

G. Post-Accident Instrumentation ~~During reactor power operation~~ A.5 ~~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 1. The post-accident MONITORING instrumentation shall be ~~functionally tested~~ CHECKED and calibrated in accordance with Table 4.2.6. A.2

THE POST-ACCIDENT MONITORING INSTRUMENTATION FOR EACH FUNCTION IN TABLE 3.2.6

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 A.3 The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

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.4

A.1

TABLE 3.2.6

MONITORING

POST-ACCIDENT INSTRUMENTATION

LA.1

	Minimum Number of Operable Instrument Channels (Note 8) A.4	Parameter	Type of Indication	Instrument Range
1.	2	Drywell Atmospheric Temperature (Notes 1 and 3)	Recorder #TR-16-19-45 (Blue) Meter #TI-16-19-30B	0-350°F 0-350°F
2.	2	DRYWELL ~~Containment~~ Pressure (Notes 1 and 3)	Meter #PI-16-19-12A Meter #PI-16-19-12B	(-15) -(+260) psig (-15) -(+260) psig
3.	2	Torus Pressure (Notes 1 and 3)	Meter #PI-16-19-36A Meter #PI-16-19-36B	(-15) -(+65) psig (-15) -(+65) psig
4.	2	Torus Water Level (Notes 1 and 3)	Meter #LI-16-19-12A Meter #LI-16-19-12B	0-25 ft. 0-25 ft.
5.	2	Torus Water Temperature (Notes 1 and 3)	Meter #TI-16-19-33A Meter #TI-16-19-33C	0-250°F 0-250°F
6.	2	Reactor Pressure (Notes 1 and 3)	Meter #PI-2-3-56A Meter #PI-2-3-56B	0-1500 psig 0-1500 psig
7.	2	Reactor Vessel Water Level (Notes 1 and 3)	Meter #LI-2-3-91A Meter #LI-2-3-91B	(-200)-0-(+200)"H_2O (-200)-0-(+200)"H_2O
8.	2	Torus Air Temperature (Notes 1 and 3)	Recorder #TR-16-19-45 (Red) Meter #TI-16-19-41	0-350°F 50-300°F
	2/valve	Safety/Relief Valve Position From Pressure Switches (Notes 1 and 3)	Lights RV-2-71(A-D) From PS-2-71-(1-3)(A-D)	Closed - Open

R.1

A.1

TABLE 3.2.6
(Cont'd)

MONITORING

POST-ACCIDENT INSTRUMENTATION

LA.1

Minimum Number of Operable Instrument Channels (Note 8) A.4	Parameter	Type of Indication	Instrument Range	
1/valve	Safety Valve Position From Acoustic Monitor (Note 5)	Meter ZI-2-1A/B	Closed - Open	R.1
9. 2	Containment Hydrogen/Oxygen Monitor (Notes 1 and 3)	Recorder SR-VG-6A (SI) Recorder SR-VG-6B (SII)	0-30% hydrogen 0-25% oxygen	LA.1
10. 2	Containment High-Range Radiation Monitor (Note 6)	Meter RM-16-19-1A/B	1 R/hr-10^7 R/hr	
1	Stack Noble Gas Effluent (Note 7)	Meter RM-17-155	0.1 - 10^7 mR/hr	R.1

A.1

ACTION

TABLE 3.2.6 NOTES

~~Note 1~~ - Within 30 days following the loss of one indication, restore the inoperable channel to an operable status or a special report to the Commission must be prepared and submitted within the subsequent 14 days, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

TABLE 3.2.6 ACTION NOTE 1.a

~~Note 2 - Deleted.~~

~~Note 3~~ - Within 7 days following the loss of both indications, restore at least one required channel to an operable status or place the reactor in a hot shutdown condition within the following 12 hours.

TABLE 3.2.6 ACTION NOTE 1.b

~~Note 4 - Deleted.~~

R.1

Note 5 - From and after the date that safety valve position from the acoustic monitor is unavailable, reactor operation may continue for 30 days provided safety valve position can be determined by monitoring safety valve discharge temperature and primary containment pressure. If after 30 days the inoperable channel has not been returned to an operable status, a special report to the Commission must be prepared and submitted within the subsequent 14 days, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

If one or both parameters are not available (i.e., safety valve discharge temperature and primary containment pressure indication) with one or more safety valve position indications from the acoustic monitor unavailable, continued reactor operation is permissible during the next seven days. In this condition, if both secondary parameters are not restored to an operable status within seven days, the reactor shall be placed in a hot shutdown condition within the following 12 hours.

~~Note 6~~ - Within 30 days following the loss of one indication, or seven days following the loss of both indications, restore the inoperable channel(s) to an operable status or a special report to the Commission must be prepared and submitted within the subsequent 14 days, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

TABLE 3.2.6 ACTION NOTE 2

Note 7 - From and after the date that this parameter is unavailable by Control Room indication, within 72 hours ensure that local sampling capability is available. If the Control Room indication is not restored within 7 days, prepare and submit a special report to the NRC within 14 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

R.1

~~Note 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 action notes may be delayed for up to 6 hours.

A.4

A.1

VYNPS

Tests and Frequencies

TABLE 4.2.6

CALIBRATION REQUIREMENTS

MONITORING

POST-ACCIDENT INSTRUMENTATION

	Parameter	Calibration	Instrument Check
1.	Drywell Atmosphere Temperature	Every 6 Months	Once Each Day
2.	Containment Pressure (Drywell)	Once/Operating Cycle	Once Each Day
3.	Torus Pressure	Once/Operating Cycle	Once Each Day
4.	Torus Water Level	Once/Operating Cycle	Once Each Day
5.	Torus Water Temperature	Every 6 Months	Once Each Day
6.	Reactor Pressure	Once/Operating Cycle	Once Each Day
7.	Reactor Vessel Water Level	Once/Operating Cycle	Once Each Day
8.	Torus Air Temperature	Every 6 Months	Once Each Day
	Safety/Relief Valve Position	Every Refueling Outage (Note 9) (a Functional Test to be performed quarterly)	Once Each Day
	Safety Valve Position	Every Refueling Outage (Note 9) (a Functional Test to be performed quarterly)	Once Each Day

R.1

A.1

VYNPS

TABLE 4.2.6

CALIBRATION REQUIREMENTS

Tests and frequencies

POST-ACCIDENT INSTRUMENTATION (Cont'd)

MONITORING

	Parameter	Calibration	Instrument Check
9.	Containment Hydrogen/Oxygen Monitor	Once/Operating Cycle	Once each day
10.	Containment High-Range Radiation Monitor	Once/Operating Cycle	Once each day
	Stack Noble Gas Effluent	Every Operating Cycle (a Functional Test to be performed quarterly)	Once each day

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.

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.6

5. Deleted.

6. Deleted.

7. Deleted.

8. Functional tests and calibrations are not required when systems are not required to be operable. A.6

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. R.1

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.6

Safety Assessment

Discussion of Changes

3.2.G and 4.2.G

Post-Accident Monitoring 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.G includes reference to CTS Table 4.2.6 for functional test and calibration requirements for post-accident monitoring instrumentation. CTS 4.2.G is revised, in proposed Surveillance Requirement (SR) 4.2.G.1, to delete the reference to functional testing and include reference to check requirements consistent with CTS Table 4.2.6. (CTS Table 4.2.6 includes Check requirements, but does not include Functional Test requirements). This change is a presentation preference and does not alter the current requirements for periodic testing of post-accident monitoring instrument Functions. Therefore, this change is considered administrative in nature.

A.3 CTS 3.2.H and 4.2.H provide requirements that apply to drywell to torus ΔP instrumentation. Changes to these CTS drywell to torus ΔP instrumentation requirements are addressed in the Safety Assessment 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.

A.4 CTS Table 3.2.6 Note 8 provides an allowance to delay entry into actions for 6 hours for the situation of a post-accident monitoring instrumentation channel inoperable solely for performance of surveillances. This allowance is moved to proposed SR 4.2.G.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 CTS 3.2.G specifies an Applicability for post-accident monitoring instrumentation of "During reactor power operation." The CTS definition of reactor power operation states "Reactor power operation is any operation with the mode switch in the Startup/Hot Standby or Run..." This change provides an explicit Applicability, in proposed Table 3.2.6 for each post-accident monitoring instrumentation Function. The specified Applicabilities, in proposed Table 3.2.6, are consistent with the CTS definition of reactor power operation (i.e., RUN and STARTUP/HOT STANDBY). 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.6 CTS Table 4.2 Notes 2, 3, 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, 3, 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 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 Table 4.2 Notes 4, 12, and 13 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 3.2.6 details relating to system design and operation (i.e., type of indication, specific instrument tag numbers, and instrument range) are unnecessary in the TS and are proposed to be relocated to the Technical Requirements Manual (TRM). Proposed Specification 3.2.G and Table 3.2.6 require the post-accident monitoring instrument Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.6 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required post-accident monitoring instrument 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.

RELOCATED SPECIFICATIONS

R.1 3.2.G/4.2.G POST-ACCIDENT INSTRUMENTATION

LCO Statement:

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.

RELOCATED SPECIFICATIONS

R.1 (continued)

Discussion:

Each individual post-accident monitoring parameter has a specific purpose; however, the general purpose for accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding per prediction, i.e. automatic safety systems are performing properly, and deviations from expected accident course are minimal.

Comparison to Deterministic Screening Criteria:

The NRC position on application of the deterministic screening criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988 from T.E. Murley (NRC) to R.F. Janecek (BWROG). The position was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the VYNPS Regulatory Guide 1.97 instruments. Those instruments meeting these criteria have remained in Technical Specifications. The instruments not meeting these criteria have been relocated from the Technical Specifications to plant controlled documents.

The following summarizes the VYNPS position for those instruments currently in Technical Specifications.

From NRC SER dated December 4, 1990, Subject: Conformance to Regulatory Guide 1.97 for Vermont Yankee Nuclear Power Station.

Type A Variables

1. Reactor Pressure
2. Reactor Vessel Level
3. Drywell Pressure
4. Drywell Temperature
5. Torus Pressure
6. Torus Water Temperature
7. Torus Water Level
8. Torus Airspace Temperature

From Regulatory Guide 1.97 and VYNPS submittal to the NRC dated October 30, 1984, NUREG-0737, Supplement 1 - Regulatory Guide 1.97, as modified by VYNPS letters dated October 25, 1985, August 11, 1987, July 28, 1988, September 1, 1989, and March 29, 1996, and NRC letter dated April 29, 1993.

RELOCATED SPECIFICATIONS

R.1 (continued)

Other Type, Category 1 Variables

Primary Containment Isolation Valve Position
Containment and Drywell Hydrogen Concentration
Containment and Drywell Oxygen Concentration
Primary Containment High Radiation

For other post-accident monitoring instrumentation currently in Technical Specifications, their loss is not risk-significant since the variable they monitored did not qualify as a Type A or Category 1 variable (one that is important to safety and needed by the operator, so that the operator can perform necessary normal actions).

Conclusion

Since the screening criteria have not been satisfied for non-Regulatory Guide 1.97 Type A or Category 1 variable instruments, their associated LCO, Actions and Surveillances may be relocated to the Technical Requirements Manual. Changes to the Technical Requirements Manual are controlled using 10 CFR 50.59. The instruments to be relocated are as follows:

1. Safety/Relief Valve Position from Pressure Switches
2. Safety Valve Position from Acoustic Monitor
3. Stack Noble Gas Effluent

No Significant Hazards Consideration

3.2.G and 4.2.G

Post-Accident Monitoring 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.

"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.

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.

TECHNICAL CHANGES – LESS RESTRICTIVE

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

References

3.2.G and 4.2.G

Post-Accident Monitoring Instrumentation

3.2.G/4.2.G REFERENCES
Post-Accident Monitoring Instrumentation

1. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.

2. NRC letter, M.B. Fairtile (NRC) to L.A. Temblay (VYNPC), "Conformance to Regulatory Guide 1.97 for Vermont Yankee Nuclear Power Station," December 4, 1990.

Proposed

Technical Specifications

3.2.H and 4.2.H

Drywell to Torus ΔP Instrumentation

3.2 LIMITING CONDITIONS FOR OPERATION

H. Deleted.

I. Recirculation Pump Trip Instrumentation

The recirculation pump trip instrumentation for each Trip Function in Table 3.2.7 shall be operable in accordance with Table 3.2.7.

J. Deleted.

4.2 SURVEILLANCE REQUIREMENTS

H. Deleted.

I. Recirculation Pump Trip Instrumentation

1. The recirculation pump trip instrumentation shall be checked, functionally tested and calibrated in accordance with Table 4.2.7.

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

2. Perform a Logic System Functional Test, including recirculation pump trip breaker actuation, of recirculation pump trip instrumentation Trip Functions once every Operating Cycle.

J. Deleted

Current

Technical Specifications

Markups

3.2.H and 4.2.H

Drywell to Torus ΔP Instrumentation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION	4.2 SURVEILLANCE REQUIREMENTS
2. If the required action and associated completion time of Specification 3.2.F.1 is not met, within the following 12 hours: a. Isolate the mechanical vacuum pump; or b. Isolate the main steam lines; or c. Place the reactor in the SHUTDOWN Mode.	3. Perform an instrument calibration, except for the radiation detectors, using a current source once every three (3) months. The trip setting shall be ≤ 3.0 times background at rated thermal power. 4. Perform an instrument calibration using a radiation source once each refueling outage. 5. Perform a logic system functional test, including mechanical vacuum pump isolation valve, once each operating cycle.
G. <u>Post-Accident Instrumentation</u> 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. <u>Post-Accident Instrumentation</u> The post-accident instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.6.
H. <u>Drywell to Torus ΔP Instrumentation</u> 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. <u>Drywell to Torus ΔP Instrumentation</u> The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

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LC.1

A.1

3.2 LIMITING CONDITIONS FOR OPERATION	4.2 SURVEILLANCE REQUIREMENTS
~~sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.~~	LC.1
I. Recirculation Pump Trip Instrumentation During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operable in accordance with Table 3.2.1.	I. Recirculation Pump Trip Instrumentation The Recirculation Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.
J. Deleted	J. Deleted
K. Degraded Grid Protective System During reactor power operation, the emergency bus undervoltage instrumentation shall be operable in accordance with Table 3.2.8.	K. Degraded Grid Protective System The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.
L. Reactor Core Isolation Cooling System Actuation When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G, the instrumentation which initiates actuation of this system shall be operable in accordance with Table 3.2.9.	L. Reactor Core Isolation Cooling System Actuation Instrumentation and Logic Systems shall be functionally tested and calibrated as indicated in Table 4.2.9.

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Safety Assessment

Discussion of Changes

3.2.H and 4.2.H

Drywell to Torus ΔP Instrumentation

SAFETY ASSESSMENT OF CHANGES
CTS: 3.2.H/4.2.H – DRYWELL TO TORUS ΔP 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

LC.1 CTS 3.2.H and 4.2.H specify requirements for the drywell to torus Δp instrumentation. This monitoring instrumentation does not necessarily relate directly to maintaining the monitored parameter (drywell to torus Δp) within limits. The ISTS do not specify indication-only equipment to be operable to support operability of a system or component or maintaining variables within limits. Control of the availability of, and necessary compensatory activities if not available, for indication and monitoring instruments are addressed by plant procedures and policies. Therefore, these requirements are to be relocated to the Technical Requirements Manual. In addition, TS 3.7.A.9/4.7.A.9, Drywell/Suppression Chamber d/p, provides regulatory control over the requirement to maintain drywell to torus Δp within limits. As a result, the details to be relocated are not required to be in the Technical Specifications to provide adequate protection of the public health and safety. Changes to the Technical Requirements Manual are controlled using 10 CFR 50.59. Not including these requirements in the TS is consistent with the ISTS.

RELOCATED SPECIFICATIONS

None

No Significant Hazards Consideration

3.2.H and 4.2.H

Drywell to Torus ΔP Instrumentation

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" LESS RESTRICTIVE CHANGES:
RELOCATION OF INSTRUMENTATION ONLY REQUIREMENTS
("LC.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 instrumentation requirements from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents. These requirements are not considered in the safety analysis. 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 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 requirements 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 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 assumption. In addition, the requirements 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 requirements 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 requirements 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 instrumentation requirements ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3.2.H/4.2.H – DRYWELL TO TORUS ΔP INSTRUMENTATION

TECHNICAL CHANGES – LESS RESTRICTIVE

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

Proposed

Technical Specifications

3.2.I and 4.2.I
Recirculation Pump Trip Instrumentation

3.2 LIMITING CONDITIONS FOR OPERATION

H. Deleted.

I. Recirculation Pump Trip Instrumentation

The recirculation pump trip instrumentation for each Trip Function in Table 3.2.7 shall be operable in accordance with Table 3.2.7.

J. Deleted.

4.2 SURVEILLANCE REQUIREMENTS

H. Deleted.

I. Recirculation Pump Trip Instrumentation

1. The recirculation pump trip instrumentation shall be checked, functionally tested and calibrated in accordance with Table 4.2.7.

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

2. Perform a Logic System Functional Test, including recirculation pump trip breaker actuation, of recirculation pump trip instrumentation Trip Functions once every Operating Cycle.

J. Deleted

Table 3.2.7 (page 1 of 1)
Recirculation Pump Trip Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Low-Low Reactor Vessel Water Level	RUN	2	Note 1	≥ 82.5 inches
2. Time Delay	RUN	2	Note 1	≤ 10 seconds
3. High Reactor Pressure	RUN	2	Note 1	≤ 1150 psig

Table 3.2.7 ACTION Notes

1. With one or more recirculation pump trip instrumentation channels inoperable, take all of the applicable Actions in Notes 1.a, 1.b and 1.c below.

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

      1) Restore the channel to operable status within 14 days; or
      2) Place the inoperable channel in trip within 14 days (not applicable for Trip Function 2 channels and not applicable if the inoperable channel is the result of an inoperable recirculation pump trip breaker).

   b. With Trip Functions 1 and 2 with recirculation pump trip capability not maintained or with Trip Function 3 with recirculation pump trip capability not maintained:

      1) Restore recirculation pump trip capability within 72 hours.

   c. With Trip Functions 1, 2 and 3 with recirculation pump trip capability not maintained:

      1) Restore recirculation pump trip capability for all but one Trip Function within 1 hour.

   If any applicable Action and associated completion time of Note 1.a, 1.b or 1.c is not met, immediately take the applicable Action of Note 2.a or 2.b.

2. a. Remove affected recirculation pump from service within the next 6 hours; or

   b. Place the plant in STARTUP/HOT STANDBY within the next 6 hours.

Table 4.2.7 (page 1 of 1)
Recirculation Pump Trip Instrumentation
Tests and Frequencies

TRIP FUNCTION	CHECK	FUNCTIONAL TEST	CALIBRATION
1. Low-Low Reactor Vessel Water Level	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
2. Time Delay	NA	NA	Every 3 Months
3. High Reactor Pressure	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle

(1) Trip unit calibration only.

Proposed

Bases

3.2.I and 4.2.I

Recirculation Pump Trip Instrumentation

BASES: 3.2.I/4.2.I RECIRCULATION PUMP TRIP INSTRUMENTATION

BACKGROUND

The Anticipated Transient Without Scram (ATWS) Prevention/Mitigation System initiates a Recirculation Pump Trip (RPT), adding negative reactivity, following events in which a scram does not but should occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Low - Low Reactor Vessel Water Level or High Reactor Pressure setpoint is reached, the reactor recirculation motor generator (RRMG) field breakers trip.

The RPT Instrumentation (Ref. 1) of the ATWS Prevention/Mitigation System includes sensors, relays, and switches that are necessary to cause initiation of an RPT. The 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 an RPT signal to the trip logic.

The RPT Instrumentation consists of two independent and identical trip systems (A and B), with two channels of High Reactor Pressure and two channels of Low - Low Reactor Vessel Water Level in each trip system. Each RPT Instrumentation trip system is a two-out-of-two logic for each Trip Function. Thus, either two Low - Low Reactor Water Level or two High Reactor Pressure signals will trip a trip system. In addition, a combination of one Low - Low Reactor Vessel Water Level signal and one High Reactor Pressure signal (in the same trip system) will trip the trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective RRMG field breakers). Each Low - Low Reactor Vessel Water Level channel output must remain below the setpoint for approximately 10 seconds for the channel output to provide an actuation signal to the associated trip system.

There is one RRMG field breaker provided for each of the two recirculation pumps for a total of two breakers. The output of each trip system is provided to both RRMG field breakers.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The RPT Instrumentation is not assumed to mitigate any accident or transient in the safety analysis. The RPT Instrumentation initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The operability of the RPT Instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions. 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

BASES: 3.2.I/4.2.I RECIRCULATION PUMP TRIP INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Trip Settings specified in Table 3.2.7. 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. Channel operability also includes the associated recirculation pump trip breakers (i.e., RRMG field breakers).

The individual Trip Functions are required to be operable in the RUN Mode to protect against common mode failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The High Reactor Pressure and Low - Low Reactor Vessel Water Level Trip Functions are required to be operable in the RUN Mode, since the reactor is producing significant power and the recirculation system could be at high flow. During this Mode, the potential exists for pressure increases or low water level, assuming an ATWS event. In the STARTUP/HOT STANDBY Mode, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the RPT Instrumentation is not necessary. In HOT SHUTDOWN and COLD SHUTDOWN, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In the REFUELING Mode, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant.

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

1, 2. Low - Low Reactor Vessel Water Level and Time Delay

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, RPT is initiated at low-low RPV water level to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and thermal power and, therefore, the rate of coolant boiloff.

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, with two channels in each trip system, are available and required to be operable to ensure that no single instrument failure can preclude an RPT from this Trip Function on a valid signal. In addition, a time delay is associated with each Low - Low Reactor Vessel Water Level channel which delays the Low - Low Reactor Vessel Water Level Trip Function channel output signal from providing input to the associated trip system. Four channels of Time Delay, with two channels in each trip system, are available and required to be operable to ensure that no single instrument failure can preclude an RPT from the Low - Low Reactor Vessel Water Level Trip Function on a valid signal.

BASES: 3.2.I/4.2.I RECIRCULATION PUMP TRIP INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low - Low Reactor Vessel Water Level Trip Setting is chosen so that RPT will not interfere with the Reactor Protection System. The Trip Setting is referenced from the top of enriched fuel. The Trip Setting of the Time Delay associated with the Low - Low Reactor Vessel Water Level Trip Function is chosen to avoid making the consequences of a loss of coolant accident more severe while ensuring the delay has an insignificant affect on the ATWS consequences.

3. High Reactor Pressure

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and thermal power, which could potentially result in fuel failure and overpressurization. The High Reactor Pressure Trip Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety valves, limits the peak RPV pressure to within the required limit.

The High Reactor Pressure signals are initiated from four pressure transmitters that monitor reactor pressure. Four channels of High Reactor Vessel Pressure, with two channels in each trip system, are available and are required to be operable to ensure that no single instrument failure can preclude an RPT from this Trip Function on a valid signal. The High Reactor Vessel Pressure Trip Setting is chosen to provide an adequate margin to the maximum allowable Reactor Coolant System pressure.

ACTIONS

Table 3.2.7 ACTION Note 1

For Trip Functions 1, 2, and 3, with one or more Trip Function channels inoperable, but with RPT trip capability for each Trip Function maintained (refer to next paragraph), the RPT instrumentation is capable of performing the intended function. However, the reliability and redundancy of the RPT Instrumentation is reduced, such that a single failure in the remaining trip system could result in the inability of the RPT Instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to operable status. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Trip Functions, and the low probability of an event requiring the initiation of RPT, 14 days is provided to restore the inoperable channel (Table 3.2.7 ACTION Note 1.a.1)). Alternately, for Trip Functions 1 and 3, the inoperable channel may be placed in trip (Table 3.2.7 ACTION Note 1.a.2)), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Inoperable channels may be placed in trip using test jacks or other permanently installed circuits. As noted in Table 3.2.7 ACTION Note 1.a.2), placing the channel in trip with no

BASES: 3.2.I/4.2.I RECIRCULATION PUMP TRIP INSTRUMENTATION

ACTIONS (continued)

further restrictions is not allowed if the inoperable channel is a Trip Function 2 channel (i.e., Time Delay Trip Function) or is the result of an inoperable breaker, since this may not adequately compensate for the inoperable Trip Function 2 channel or inoperable breaker (e.g., the breaker may be inoperable such that it will not open), as applicable. 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 RPT), or if the inoperable channel is the result of an inoperable breaker, Table 3.2.7 ACTION Note 2 must be entered and its required Actions taken.

Table 3.2.7 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 the Trip Function 1 and 2 not maintaining RPT trip capability or Trip Function 3 not maintaining RPT trip capability. A Trip Function is considered to be maintaining RPT trip capability when sufficient channels are operable or in trip such that the RPT Instrumentation will generate a trip signal from the given Trip Function in either of the two trip systems on a valid signal, and both recirculation pumps can be tripped. For Trip Functions 1 and 2, this requires two channels of each Trip Function in the same trip system to be operable or in trip and the RRMG field breakers to be operable or in trip. For Trip Function 3, this requires two channels in the same trip system to be operable or in trip and the RRMG field breakers to be operable or in trip. The 72 hour Completion Time is sufficient for the operator to take corrective action (e.g., restoration or tripping of channels) and takes into account the likelihood of an event requiring actuation of the RPT instrumentation during this period and that Trip Functions 1 and 2 or Trip Function 3 still maintain RPT trip capability.

Table 3.2.7 ACTION Note 1.c is intended to ensure that appropriate Actions are taken if multiple, inoperable, untripped channels within Trip Functions 1, 2, and 3 result in Trip Functions 1, 2, and 3 not maintaining RPT trip capability. The description of a Trip Function maintaining RPT trip capability is discussed in the paragraph above. The 1 hour Completion Time for restoring all but one of the Trip Functions is sufficient for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of the RPT Instrumentation during this period.

Table 3.2.7 ACTION Note 2

With any required Action and associated completion time 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 STARTUP/HOT STANDBY within 6 hours (Table 3.2.7 ACTION Note 2.b).

Alternately, the associated recirculation pump may be removed from service since this performs the intended function of the instrumentation (Table 3.2.7 ACTION Note 2.a). The allowed Completion Time of 6 hours is reasonable, based on operating experience, both to reach STARTUP/HOT STANDBY from full power conditions and to remove a recirculation pump from service in an orderly manner and without challenging plant systems.

BASES: 3.2.I/4.2.I RECIRCULATION PUMP TRIP INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.I.1

As indicated in Surveillance Requirement 4.2.I.1, RPT Instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.7. Table 4.2.7 identifies, for each Trip Function, the applicable Surveillance Requirements.

Surveillance Requirement 4.2.I.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 recirculation pump trip 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. 2) 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 recirculation pumps will trip when necessary.

Surveillance Requirement 4.2.I.2

The Logic System Functional Test demonstrates the operability of the required initiation logic and simulated automatic operation for a specific channel. A system functional test of the recirculation pump trip breakers (i.e., RRMG field breakers) is included in this Surveillance to provide complete testing of the assumed safety function. Therefore, if an RRMG field breaker is incapable of operating, the associated instrument channel(s) 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.

Table 4.2.7, Check

Performance of an Instrument Check once per day, for Trip Functions 1 and 3, 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

BASES: 3.2.I/4.2.I RECIRCULATION PUMP TRIP INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (continued)

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 4.2.7, Functional Test

For Trip Functions 1 and 3, 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. For Trip Functions 1 and 3, the Frequency of "Every 3 Months" is based on the reliability analysis of Reference 2.

Table 4.2.7, Calibration

For Trip Functions 1, 2, and 3, 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 3, 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 2 and the time interval assumption for trip unit calibration used in the associated setpoint calculation.

REFERENCES

1. UFSAR, Section 7.18.

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

Current

Technical Specifications

Markups

3.2.I and 4.2.I

Recirculation Pump Trip Instrumentation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.

A.2

I. Recirculation Pump Trip Instrumentation

A.3

During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operable in accordance with Table 3.2.7

FOR EACH TRIP FUNCTION IN TABLE 3.2.7

J. Deleted

K. Degraded Grid Protective System

During reactor power operation, the emergency bus undervoltage instrumentation shall be operable in accordance with Table 3.2.8.

L. Reactor Core Isolation Cooling System Actuation

When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G, the instrumentation which initiates actuation of this system shall be operable in accordance with Table 3.2.9.

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4.2 SURVEILLANCE REQUIREMENTS

I. Recirculation Pump Trip Instrumentation

A.4

1. The Recirculation Pump Trip Instrumentation shall be CHECKED, functionally tested and calibrated in accordance with Table 4.2.7

J. Deleted

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.

L. Reactor Core Isolation Cooling System Actuation

Instrumentation and Logic Systems shall be functionally tested and calibrated as indicated in Table 4.2.9.

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 PROVIDED THE ASSOCIATED TRIP FUNCTION MAINTAINS RECIRCULATION PUMP TRIP CAPABILITY.

A.5

A.1

TABLE 3.2 ~~(1)~~ 7
~~(Cont'd)~~

RECIRCULATION PUMP TRIP ~~ACTUATION~~ INSTRUMENTATION

REQUIRED

Actions when Required Channels are Inoperable

LA.2

Recirculation Pump Trip ~~- A & B (Note 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.	2	~~(Note 8)~~ A.5	Low-Low Reactor Vessel Water Level ~~(LM-2-3-68(A-D))~~	≥ 6' 10.5" ~~above top of enriched fuel~~ LA.3	Note ~~(8)~~ 1
3.	2	~~(Note 8)~~	High Reactor Pressure ~~(PM-2-3-54(A-D))~~ LA.1	≤ 1150 psig	Note ~~(8)~~ 1
2.	2	~~(Note 8)~~	Time Delay ~~(2-3-68(A-D)(X))~~	≤ 10 seconds	Note ~~(8)~~ 1
	~~1~~		~~Trip Systems Logic~~		~~Note 2~~ L.1

A.1

TABLE 3.2.1 NOTES (ACTION)

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". LA.2

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. LA.4 L.1

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.

5. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply. A.6

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. A.5 A.6

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.

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

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

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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A.1

7 ACTION

TABLE 3.2.7 NOTES (Cont'd)

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18. With one or more channels inoperable for ADS:

A. Within one hour from discovery of loss of ADS initiation capability in one trip system, declare ADS inoperable, and
B. Within 96 hours from discovery of an inoperable channel concurrent with HPCI or RCIC System inoperable, restore channel to operable status, and
C. Within 8 days, restore channel to operable status.
D. If required actions and associated completion times of actions A, B or C are not met, immediately declare ADS inoperable.

Table 3.2.7 Action Note 1 — 19. With one or more channels inoperable for Recirculation Pump Trip: M.1

Table 3.2.7 Action Note 1.c — A. Within one hour from discovery of loss of Recirculation Pump Trip capability restore all but one one Trip Function or remove the associated recirculation pump from service in 6 hours or be in Startup/Hot Standby in 6 hours.

Table 3.2.7 Action Note 3 — B. Within 14 days from discovery of an inoperable channel, restore channel to operable status or place in trip, and

Table 3.2.7 Action Note 1.a — C. Within 72 hours from discovery of ~~one trip function capability~~ not maintained, restore trip function to operable status and,

Table 3.2.7 Action Notes 1 & 3 — D. If required actions and associated completion times of actions A, B or C are not met, immediately remove the associated recirculation pump from service in 6 hours or be in Startup/Hot Standby in 6 hours.

(not applicable for Trip Function 2 channels and not applicable if inoperable channel is the result of an inoperable recirculation pump trip breaker) M.2

Trip Functions 1 and 2 RECIRCULATION PUMP with trip capability not maintained or WITH Trip Function 3 with RECIRCULATION PUMP trip capability M.1

A.1

TABLE 3.2.7

(Table 3.2.7 was intentionally deleted from the Technical Specifications)

<replace with proposed Table 3.2.7>

A.1

TABLE 4.2.~~1~~ 7
~~(Cont'd)~~

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

~~EMERGENCY CORE~~ COOLING ~~ACTUATION~~ INSTRUMENTATION

Recirculation Pump Trip ~~Actuation System~~

	Trip Function	Functional Test ~~(8)~~ A.8	Calibration ~~(8)~~ A.8	Instrument Check
1.	Low-Low Reactor Vessel Water Level	Every Three Months ~~(Note 4)~~ A.9	Once/Operating Cycle Every 3 Months, (1) M.3	Once Each Day
3.	High Reactor Pressure	Every Three Months ~~(Note 4)~~	Once/Operating Cycle	Once Each Day
SR4.2.1.2	Trip System Logic	Once/Operating Cycle	Once/Operating Cycle A.7	-- M.5
2.	Time Delay		Every 3 Months	
	(1) Trip unit calibration only			M.3
	including recirculation trip breaker actuation			M.4

A.1

TABLE 4.2.7

(Table 4.2.7 was intentionally deleted from the Technical Specifications)

<replace with proposed Table 4.2.7>

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.10

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.9

5. ~~Deleted.~~

6. ~~Deleted.~~

7. ~~Deleted.~~

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

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.10

Safety Assessment

Discussion of Changes

3.2.I and 4.2.I

Recirculation Pump Trip Instrumentation

SAFETY ASSESSMENT OF CHANGES
TS 3.2.I/4.2.I – RECIRCULATION PUMP TRIP 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.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.

A.3 CTS 3.2.I specifies an Applicability for recirculation pump trip instrumentation of "During reactor power operation." The CTS definition of reactor power operation states "Reactor power operation is any operation with the mode switch in the Startup/Hot Standby or Run..." This change provides an explicit Applicability, in proposed Table 3.2.I for each recirculation pump instrumentation Trip Function. The specified Applicabilities, in proposed Table 3.2.6 are consistent with the CTS definition of reactor power operation as modified by the CTS Table 3.2.I Note 19 actions to exit the applicability by placing the plant in Startup/Hot Standby (i.e., Run). 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.4 CTS 4.2.I includes reference to CTS Table 4.2.1 for functional test and calibration requirements for recirculation pump trip instrumentation. CTS 4.2.I is revised, in proposed Surveillance Requirement (SR) 4.2.I.1, to also include reference to check requirements consistent with CTS Table 4.2.1. This change is a presentation preference and does not alter the current requirements to periodically perform checks of certain recirculation pump trip instrument trip functions. Therefore, this change is considered administrative in nature.

A.5 CTS Table 3.2.1 Note 8 provides an allowance to delay entry into actions for 6 hours for the situation of a recirculation pump trip instrumentation channel inoperable solely for performance of surveillances. This allowance is moved to proposed SR 4.2.I.1. This change does not involve a technical change, but is only a difference of presentation preference. Therefore, this change is considered administrative.

ADMINISTRATIVE (continued)

A.6 CTS Table 3.2.1 Notes 3, 4, 5, 6, and a portion of Note 8 provide requirements related to Emergency Core Cooling System (ECCS) instrumentation. The ECCS instrumentation is located in proposed Specifications 3.2.A and 4.2.A. Therefore, the requirements of CTS Table 3.2.1 Notes 3, 4, 5, 6, and the applicable portion of Note 8 are physically moved and changes addressed in 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.

A.7 CTS Table 4.2.1 includes a requirement to perform a calibration of recirculation pump trip instrumentation Trip System Logic once per Operating Cycle. Similar to other calibration requirements of Trip System Logic in the VYNPS CTS, the intent of this requirement is to perform a calibration of time delays necessary for proper functioning of the trip system. In proposed Table 4.2.7, this requirement is reflected with explicit requirements to perform periodic calibrations of the required recirculation pump trip instrumentation time delays (i.e., proposed Table 4.2.7 Trip Function 2, Time Delay). 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 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.9 CTS Table 4.2.1 includes Functional Test requirements for the recirculation pump trip Low – Low Reactor Vessel Water Level and High Reactor Pressure 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 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.

ADMINISTRATIVE (continued)

A.10 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 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 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 Table 4.2 Notes 12 and 13 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Table 3.2.1 Note 19.A provides actions for the condition of a loss of recirculation pump trip capability from all recirculation pump trip instrumentation trip functions, except Trip System Logic. In the event all applicable recirculation pump trip instrumentation trip functions lose trip capability, CTS Table 3.2.1 Note 19.A only requires one trip function to be restored within one hour. The intent of this action was to ensure that in the event of a loss of recirculation pump trip capability from both the low reactor vessel water level function and the high reactor pressure function, recirculation pump trip capability for at least one of these two functions would be restored within one hour to support continued operation in that condition for 72 hours. However, CTS Table 3.2.1 includes a Low – Low Reactor Vessel Water Level Trip Function (proposed Table 3.2.1 Trip Function 1); a Time Delay Trip Function (proposed Table 3.2.1 Trip Function 2), which supports the operability of the Low – Low Reactor Vessel Water Level Trip Function; and High Reactor Pressure Trip Function (proposed Trip Function 3) for which CTS Table Note 19 is applicable. As a result, if all three applicable Trip Functions lose trip capability and only one Trip Function is restored, it is still possible that a loss of recirculation pump trip capability would exist for both the low reactor vessel water level function and the high drywell pressure function (e.g., in the case where only trip capability of the Time Delay Trip Function is restored) and operation would be allowed to continue for 72 hours in this condition. Therefore, in the event all applicable recirculation pump trip instrumentation trip functions lose trip capability. CTS Table 3.2.1 Note 19.A (proposed Table 3.2.7 Action Note 1.c) is revised to require restoration of trip capability for all but one Trip Function within one hour. A corresponding change is also made to CTS Table 3.2.1 Note 19.C. CTS Table 3.2.1 Note 19.C states "Within 72 hours from discovery of discovery of one trip function capability not

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 (continued) maintained..." The CTS Table 3.2.1 Note 19.C reference to "one trip function capability not maintained" is changed to "Trip Functions 1 and 2 with trip capability not maintained or Trip Function 3 with trip capability not maintained in proposed Table 3.2.7 Action Note 1.b. This change represents an additional restriction on plant operation to ensure continued operation with a loss of recirculation pump trip capability from both the low reactor vessel water level and high reactor pressure functions is not allowed for longer than one hour. The change is consistent with the intent of the ISTS (the ISTS does not include a time delay for this type of recirculation pump trip instrumentation).

M.2 CTS Table 3.2.1 Note 19.B requires inoperable recirculation pump trip instrumentation channels to be restored or placed in the tripped condition. Proposed Table 3.2.7 Action Note 1.a provides the same alternative actions as CTS Table 3.2.1 Note 19.B but includes a limitation on the use of the action to place the inoperable channels in trip. Proposed Table 3.2.7 Action Note 1.a precludes the use of the action to trip the inoperable channel if the inoperability is associated with Trip Function 2 (i.e., Time Delay) channels if the inoperability is the result of an inoperable recirculation pump trip breaker. These restrictions are added since (1) placing the Time Delay Trip Function channels in trip (as allowed by CTS Table 3.2.1 Note 19.B) does not perform the intended function (providing a time delay for actuation after certain conditions are satisfied) and could make the consequences of a postulated LOCA more severe, and (2) with the channels inoperable due to an inoperable breaker, tripping the affected channels may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). This change represents an additional restriction on plant operation by requiring the channels in these conditions to be restored to operable status rather than tripped. The change is consistent with the ISTS.

M.3 CTS Table 4.2.1 does not include explicit requirements to calibrate trip units. Proposed Table 4.2.7 requires calibration of the trip units of the following Trip Functions every 3 months: Low - Low Reactor Vessel Water Level (proposed Table 4.2.7 Trip Function 1) and High Reactor Pressure (proposed Table 4.2.7 Trip Function 3). The trip units of these Trip Functions are currently required by CTS Table 4.2.1 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 - MORE RESTRICTIVE (continued)

M.4 CTS Table 3.2.1 requires a Functional Test of the recirculation pump trip instrumentation Trip System Logic. The CTS definition of Logic System Functional Test (CTS 1.0.H) requires where possible for the action during the test to go to completion and actuate the end device (i.e., pumps will be started and valves will be opened). For the recirculation pump trip instrumentation, actuation of the end device would require actuation of the recirculation pump trip breakers. In proposed SR 4.2.I.2, the Logic System Functional Test of the recirculation pump trip instrumentation Trip Functions explicitly requires the actuation of the recirculation pump trip breakers to be included in the test. This change represents an additional restriction on plant operation necessary to ensure complete testing of the safety function. This change is consistent with the ISTS.

M.5 CTS Table 4.2.1 includes a requirement to perform a calibration of recirculation pump trip instrumentation Trip System Logic once per Operating Cycle. Similar to other calibration requirements of Trip System Logic in the VYNPS CTS, the intent of this requirement is to perform a calibration of time delays necessary for proper functioning of the trip system once per Operating Cycle. In proposed Table 4.2.7, this requirement made more restrictive and is reflected with explicit requirements to perform calibrations of the required recirculation pump trip instrumentation time delays (i.e., proposed Table 4.2.7 Trip Function 2, Time Delay) "Every 3 Months." This change is necessary to ensure consistency with assumptions regarding the calibration frequency of these time delays used in the associated setpoint calculations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 3.2.1 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.I and Table 3.2.7 require the recirculation pump trip instrumentation Trip Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.7 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required recirculation pump trip instrumentation Trip Functions is 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 and associated Note 1 contain design and operational details of the ECCS and RPT instrumentation (i.e., nomenclature for each of the subsystems, that each of the two Core Spray, LPCI and RPT, subsystems are initiated and controlled by a trip system, and that subsystem "B is identical to subsystem "A"). These details are not necessary to ensure the operability of ECCS and RPT instrumentation. Therefore, the information in this note is to be relocated to Specifications 3.2.A and 3.2.I Bases, as applicable, and reference

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (continued) to this information is deleted from VYNPS TS. The requirements of Specifications 3.2.A and 3.2.I and the associated Surveillance Requirements for the ECCS and RPT 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.3 The Trip Setting associated with reactor vessel water level trip function (proposed Table 3.2.7 Trip Function 1) 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 recirculation pump trip instrumentation. Operability requirements are adequately addressed in proposed Specification 3.2.I, Table 3.2.7 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 The details in the CTS Table 3.2.1 Note 2, relating to the method used for placing channels in trip, are to be relocated to Specification 3.2.I Bases. The requirements of proposed Table 3.2.7 ACTIONS Notes ensure inoperable channels are placed in trip or the unit is placed in a non-applicable Mode or condition, as appropriate. As a result, the relocated details in the CTS Table 3.2.1 Note 2 are not necessary for ensuring the appropriate actions are taken in the event of inoperable recirculation pump trip instrumentation channels. 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 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 CTS Table 3.2.1 includes requirements for Trip System Logic associated with the recirculation pump trip instrumentation Trip Functions. The Trip System Logic is considered part of the recirculation pump trip instrumentation Trip Functions and the requirements for the associated Trip System Logic to be operable are encompassed by the definition of operable. Therefore, the CTS Table 3.2.1 listing of Trip System Logic as a separate Trip Function is unnecessary and is deleted. With the deletion of separate Trip System Logic Trip Function, the actions associated with inoperable Trip System Logic (CTS Table 3.2.1 Note 2) will now be governed by the actions for the individual proposed Table 3.2.7 recirculation pump trip instrumentation Trip Functions. These proposed Table 3.2.7 Action Notes are less restrictive than the CTS Table 3.2.1 Note 2 actions. However, the proposed actions will ensure, in the event of inoperabilities, that consistent actions are applied to both recirculation pump trip instrumentation Trip Functions and their associated

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (continued) Trip System Logic for the same level of degradation. This change is acceptable, since the allowed outage times of the proposed Table 3.2.7 Action Notes will limit operation to within the bounds of the applicable analysis, i.e., GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Outage Times For Selected Instrumentation Technical Specifications," December 1992. Application of this analysis to the VYNPS recirculation pump trip instrumentation Trip Functions, including the associated Trip System Logic, 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.I and 4.2.I

Recirculation Pump Trip 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 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.

"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.I/4.2.I – RECIRCULATION PUMP TRIP 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 relax the actions when one or more channels of recirculation pump trip instrumentation are inoperable due to inoperable Trip System Logic. The recirculation pump trip 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 recirculation pump trip 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 recirculation pump trip 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 recirculation pump trip 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.I and 4.2.I

Recirculation Pump Trip Instrumentation

3.2.I/4.2.I REFERENCES
Recirculation Pump Trip Instrumentation

1. UFSAR, Section 7.18.

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

Proposed

Technical Specifications

3.2.K and 4.2.K

Degraded Grid Protective System

3.2 LIMITING CONDITIONS FOR OPERATION

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation for each Trip Function in Table 3.2.8 shall be operable in accordance with Table 3.2.8.

4.2 SURVEILLANCE REQUIREMENTS

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.

Table 3.2.8 (page 1 of 1)
Degraded Grid Protective System Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER BUS	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Degraded Bus Voltage				
a. Voltage	(1)	2	Note 1	≥ 3660 volts and ≤ 3740 volts
b. Time Delay	(1)	1	Note 2	≥ 9 seconds and ≤ 11 seconds

(1) When the associated diesel generator is required to be operable.

Table 3.2.8 ACTION Notes

1. With one or more required Degraded Bus Voltage Trip Function 1 channels inoperable:

   a. Place the inoperable channel in trip within 1 hour.

   If the Action and associated completion time of Note 1.a are not met, immediately declare the associated diesel generator inoperable.

2. With one or more required Degraded Bus Voltage Trip Function 2 channels inoperable:

   a. Restore the inoperable channel to operable status within 1 hour.

   If the Action and associated completion time of Note 2.a are not met, immediately declare the associated diesel generator inoperable.

Table 4.2.8 (page 1 of 1)
Degraded Grid Protective System Instrumentation
Tests and Frequencies

TRIP FUNCTION	FUNCTIONAL TEST	CALIBRATION
1. Degraded Bus Voltage		
a. Voltage	(1)	Once/Operating Cycle
b. Time Delay	(1)	Once/Operating Cycle

(1) Separate Functional Tests are not required for this Trip Function. Trip Function operability is demonstrated during Trip Function Calibration and integrated ECCS tests performed once per Operating Cycle.

Proposed

Bases

3.2.K and 4.2.K

Degraded Grid Protective System

BASES: 3.2.K/4.2.K DEGRADED GRID PROTECTIVE SYSTEM INSTRUMENTATION

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The Degraded Grid Protective System instrumentation monitors the 4.16 kV emergency buses. Offsite power is the preferred source of power for the 4.16 kV emergency buses. If the monitors determine that insufficient voltage is available and an ECCS initiation signal is present, the buses are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4.16 kV emergency bus has its own independent Degraded Grid Protective System instrumentation and associated trip logic. The voltage for each bus is monitored for degraded voltage.

The Degraded Bus Voltage - Voltage Trip Function is monitored by two undervoltage relays for each 4.16 kV emergency bus, whose outputs are arranged in a two-out-of-two logic configuration (Ref. 1). For the Degraded Bus Voltage - Time Delay Trip Function, two channels for each 4.16 kV emergency bus are provided. However, only one Degraded Bus Voltage - Time Delay channel per bus is dedicated to the DG start function. The other Degraded Bus Voltage - Time Delay channel per bus is dedicated to a control room annunciator function from which manual action is taken for degraded grid protection when an accident signal is not present. The Degraded Bus Voltage - Time Delay Trip Function is nominally adjusted to 10 seconds since this would be indicative of a sustained degraded voltage condition. When a Degraded Bus Voltage - Voltage Trip Function setpoint has been exceeded and persists for nominally ten seconds, either one of the two Degraded Bus Voltage - Voltage Trip Function channels on an associated 4.16 kV emergency bus will actuate a control room annunciator to alert the operator of the degraded voltage condition. (However, the associated control room annunciator is not subject to the requirements of this Technical Specification) If this sustained degraded voltage condition occurs coincident with a loss of coolant accident (LOCA), the 4.16 kV emergency buses are disconnected from the offsite power sources and connected to the DG power sources. If the sustained degraded voltage condition does not exist at the time of a LOCA, 4.16 kV emergency buses are not disconnected from the offsite power sources and the ECCS loads will start immediately from their normal supplies.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The degraded grid protection assures the ECCS loads and other assumed systems powered from the DGs are powered from the offsite power system as long as offsite power system voltage is within an acceptable value and it assures that loads powered from the DGs when bus voltage is insufficient for continuous operation of the connected loads. The Degraded Grid Protective System instrumentation is required for Engineered Safety Features to function in any accident with a degradation or loss of offsite power. The required channels of Degraded Grid Protective System instrumentation ensure that the ECCS and other assumed systems powered from the DGs, provide plant protection in the event of any of the Reference 2 and 3 analyzed accidents in which a loss of offsite power is assumed. The initiation of the DGs on degradation or loss of offsite power, and subsequent initiation of the ECCS, ensure that the requirements of 10 CFR 50.46 are met.

BASES: 3.2.K/4.2.K DEGRADED GRID PROTECTIVE SYSTEM INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Accident analyses credit the loading of the DGs based on the loss of offsite power coincident with a loss of coolant accident (LOCA). The diesel starting and loading times have been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power.

The Degraded Grid Protective System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The operability of the Degraded Grid Protective System instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions. 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 3.2.8. 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 specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below for the Degraded Grid Protective System instrumentation Trip Functions.

1.a, 1.b. Degraded Bus Voltage - Voltage and Degraded Bus Voltage - Time Delay

A reduced voltage condition on a 4.16 kV emergency bus indicates that, while offsite power may not be completely lost to the respective emergency bus, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the bus is transferred from offsite power to onsite DG power when the voltage on the bus drops below the Degraded Bus Voltage - Voltage Trip Function trip setpoint, is sustained in a degraded condition for approximately 10 seconds and a LOCA condition exists (as indicated by ECCS Low - Low Reactor Vessel Water Level or High Drywell Pressure Trip Function signals). This ensures that adequate power will be available to the required equipment.

The Degraded Bus Voltage Trip Settings are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Trip Settings are long enough to provide time for voltage on the station emergency bus to recover from transients such as motor starts or fault clearing, but short enough to ensure that the operating equipment is not damaged by low voltage.

Two channels of Degraded Bus Voltage - Voltage Trip Function and one channel of Degraded Bus Voltage - Time Delay Trip Function per associated bus are required to be operable when the associated DG is required to be operable to ensure that no single instrument failure can preclude the DG function.

BASES: 3.2.K/4.2.K DEGRADED GRID PROTECTIVE SYSTEM INSTRUMENTATION

ACTIONS

Table 3.2.8 ACTION Note 1

With one or more required channels of the Degraded Bus Voltage - Voltage Trip Function inoperable, the Trip Function is not capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable channel to operable status. 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 3.2.8 ACTION Note 1.a. The inoperable channel may be tripped using test jacks or other permanently installed circuits. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the Degraded Grid Protective System instrumentation), and allow operation to continue. 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.

If the Action and associated completion time of Table 3.2.8 ACTION Note 1.a are not met, the associated Trip Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable LCO and required Actions of the DG Technical Specifications, which provide appropriate actions for the inoperable DG(s).

Table 3.2.8 ACTION Note 2

With one or more required channels of the Degraded Bus Voltage - Time Delay Trip Function inoperable, the Trip Function is not capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable channel to operable status (Table 3.2.8 ACTION Note 2.a). Table 3.2.8 ACTION Note 2.a. 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. 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 of channels.

If the Action and associated completion time of Table 3.2.8 ACTION Note 2.a are not met, the associated Trip Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable LCO and required Actions of the DG Technical Specifications, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.K.1

As indicated in Surveillance Requirement 4.2.K.1, Degraded Grid Protective System instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.8. Table 4.2.8 identifies, for each Trip Function, the applicable Surveillance Requirements.

SURVEILLANCE REQUIREMENTS (continued)

Table 4.2.8, Functional Test

For Trip Functions 1.a and 1.b, as indicated in Table 4.2.8 Footnote (1), separate Functional Tests are not required since Trip Function operability is demonstrated during the Trip Function Calibration and integrated ECCS test performed once per Operating Cycle. For the Trip Function Calibration, the "once per Operating Cycle" Frequency is based upon the time interval assumptions for calibration used in the determination of the magnitude of equipment drift in the associated setpoint analyses. For the integrated ECCS test, the "once per Operating Cycle" Frequency 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 integrated ECCS test when performed at the specified Frequency.

Table 4.2.8, 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.

REFERENCES

1. UFSAR, Section 8.5.3.

2. UFSAR, Section 6.5.

3. UFSAR, Chapter 14.

Current

Technical Specifications

Markups

3.2.K and 4.2.K

Degraded Grid Protective System

A.1

3.2 LIMITING CONDITIONS FOR OPERATION	4.2 SURVEILLANCE REQUIREMENTS
sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.	A.2
	<MOVE TO SEPARATE PAGE>
I. Recirculation Pump Trip Instrumentation During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operable in accordance with Table 3.2.1.	I. Recirculation Pump Trip Instrumentation The Recirculation Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.
J. Deleted	J. Deleted
K. Degraded Grid Protective System A.3 M.1 During reactor power operation, the emergency bus undervoltage instrumentation shall be operable in accordance with Table 3.2.8. FOR EACH TRIP FUNCTION IN TABLE 3.2.8	K. Degraded Grid Protective System The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.
L. Reactor Core Isolation Cooling System Actuation When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G, the instrumentation which initiates actuation of this system shall be operable in accordance with Table 3.2.9.	L. Reactor Core Isolation Cooling System Actuation Instrumentation and Logic Systems shall be functionally tested and calibrated as indicated in Table 4.2.9.
	<MOVE TO SEPARATE PAGE>

A.1

TABLE 3.2.8

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

Degraded Grid Protective SYSTEM

	~~Minimum Number of Operable Instruments~~ REQUIRED CHANNELS PER BUS	TRIP FUNCTION ~~Parameter~~	Trip Setting	~~Required Action~~ ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE
1.a	2 per bus	Degraded Bus Voltage - Voltage (27/3Z, 27/3W, 27/4Z, 27/4W) LA.1	~~3,700 volts ± 40 volts~~ ≥ 3660 VOLTS AND ≤ 3740 VOLTS	Note 1
1.b	~~2~~ 1 per bus LC.1	Degraded Bus Voltage - Time Delay (62/3W, 62/3Z, 62/4W, 62/4Z) LA.1 LC.1	~~10 seconds ± 1 second~~ ≥ 9 SECONDS AND ≤ 11 SECONDS	Note 2

TABLE 3.2.8 NOTES Action

1. 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 within one hour. LA.2 Restore inoperable channel to operable status within 1 hour L.1

2. If the minimum number of operable instrument channels are not available, ~~reactor power operation is permissible for only 7 successive days unless the system is sooner made operable.~~ If the Action and associated completion times OF NOTE 2.a are not met, immediately declare the associated diesel generator inoperable. L.1 L.2

A.1

TABLE 4.2.8

TESTS AND FREQUENCIES

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

Degraded Grid Protective SYSTEM

	Trip System	Functional Test	Calibration (8) A.4
1.a and 1.b	Degraded Bus Voltage	See Note (10) 1	Once/Operating Cycle

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.

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.5

5. Deleted.

6. Deleted.

7. Deleted.

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

A.4

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.

A.5

FOOTNOTE (1) ~~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.5

Safety Assessment

Discussion of Changes

3.2.K and 4.2.K

Degraded Grid Protective System

SAFETY ASSESSMENT OF CHANGES
TS 3.2.K/4.2.K – DEGRADED GRID PROTECTIVE SYSTEM 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.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.

A.3 CTS 3.2.K specifies an Applicability for Degraded Grid Protective System instrumentation of "During reactor power operation." This change provides an explicit Applicability, in proposed Table 3.2.8 for each Degraded Grid Protective instrumentation Trip Function. The specified Applicability, in proposed Table 3.2.8, is consistent with the Modes and conditions specified in CTS 3.2.K, except as provided and justified in change M.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.4 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.5 CTS Table 4.2 Notes 2, 3 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, 3 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

ADMINISTRATIVE (continued)

A.5 (continued) technical change, but is only a difference of presentation preference and is considered administrative. CTS 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 Table 4.2 Notes 4, 12 and 13 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.2.K requires the Degraded Grid Protective System instrumentation to be operable only during reactor power operation. In proposed Table 3.2.8, the Applicability of the Degraded Grid Protective instrumentation Trip Functions is expanded in Footnote (1) to, "When the associated diesel generator is required to be operable." The VYNPS CTS Applicability for diesel generators, which are supported by the Degraded Grid Protective instrumentation, includes more than just "during power operation." For example, CTS 3.5.H.4 (which provides low pressure Emergency Core Cooling System (ECCS) requirements) requires a diesel generator to be operable during Refuel or Cold Shutdown when operations with the potential for draining the reactor vessel are being performed and CTS 3.7.B.1.b (which provides Standby Gas Treatment System requirements) requires diesel generators to operable under certain conditions during Refuel or Cold Shutdown when secondary containment integrity is required. This change represents an additional restriction on plant operation necessary to ensure that the ECCS and other assumed systems powered by the diesel generators remain capable of providing plant protection during the conditions when the diesel generators are required to be operable.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 3.2.8 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.K and Table 3.2.8 require Degraded Grid Protective instrumentation Trip Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.8 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required Degraded Grid Protective instrumentation Trip Functions is 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The details in the CTS Table 3.2.8 Note 1, relating to the method used for placing channels in trip, are to be relocated to Specification 3.2.K Bases. The requirements of proposed Table 3.2.8 Action Notes are adequate to ensure inoperable channels are placed in trip. As a result, the relocated details in the CTS Table 3.2.8 Note 1 are not necessary for ensuring the appropriate actions are taken in the event of inoperable Degraded Grid Protective instrumentation channels. 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 TS Bases are controlled by the provisions of 10 CFR 50.59. Not including these details in TS is consistent with the ISTS.

LC.1 CTS Table 3.2.8 specifies requirements for the operability of the Degraded Bus Voltage alarm instrumentation by requiring that relays 62/3Z and 62/4Z be operable. These relays provide an alarm only function. The ISTS do not specify alarm-only equipment to be operable to support operability of a system or component or maintaining variables within limits. Control of the availability of, and necessary compensatory activities if not available, for indication and monitoring instruments are addressed by plant procedures and policies. Therefore, these requirements are to be relocated to the Technical Requirements Manual. As such, the details to be relocated are not required to be in the Technical Specifications to provide adequate protection of the public health and safety. Changes to the Technical Requirements Manual are controlled using 10 CFR 50.59. Not including these requirements in the TS is consistent with the ISTS.

"Specific"

L.1 When one or more Degraded Bus Voltage – Time Delay channels are inoperable, CTS Table 3.2.8, Note 2 limits operation to 7 days, but does not include explicit actions to restore the inoperable channels. In proposed Table 3.2.8 Action Note 2, an explicit requirement is provided to restore the inoperable channels within 1 hour prior to requiring the associated diesel generators to be declared inoperable. Since the applicable VYNPS diesel generator TS allows operation for up to 7 days with an inoperable diesel generator, the total time allowed for the plant to remain in reactor power operation with an inoperable Degraded Voltage – Time Delay channel is extended from 7 days to 7 days + 1 hour. The 1 hour time period for this condition is provided to attempt to evaluate and repair any discovered inoperabilities prior to declaring the associated diesel generator inoperable. This 1 hour time period is considered to be acceptable because it minimizes risk while providing time for restoration or tripping of channels. For the proposed Table 3.2.8 Action Note 1 action to declare the associated diesel generator inoperable, this instrumentation, in conjunction with an ECCS initiation signal, provides a start signal for the diesel generators (i.e., it supports diesel generator operability). Therefore, when this instrumentation is inoperable and not restored within the required time period, the appropriate action is to declare the diesel generator inoperable. This is acceptable since the VYNPS TS requirements for diesel generators establish appropriate restrictions and compensatory measures for an inoperable diesel generator. The change is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 CTS Table 3.2.8 Note 1, in the event of one or more inoperable Degraded Bus Voltage – Voltage channels, requires the channel to be tripped within one hour, but does not provide direction regarding actions to take if the associated channels are not tripped. As such, a shutdown of the reactor would be required in accordance with 10 CFR 50.36(c)(2). Under these conditions, proposed Table 3.2.8 Action Note 1 provides actions to declare the associated diesel generator inoperable, which results in entering and taking the appropriate actions in the associated diesel generator TS if a channel is not tripped or restored within 1 hour. Since this instrumentation, in conjunction with an ECCS initiation signal, provides a start signal for the diesel generators (i.e., it supports diesel generator operability), the appropriate action, in this condition, would be to declare the diesel generator inoperable. The current requirements of CTS Table 3.2.8 Note 1 are overly restrictive, in that if the diesel were inoperable for other reasons, a 7 day restoration time is provided; yet currently if an instrument is inoperable and not tripped within one hour, but the diesel is otherwise fully operable, an immediate shutdown is required. This change is acceptable since the VYNPS TS requirements for diesel generators establish appropriate restrictions and compensatory measures for an inoperable diesel generator. The change is consistent with the ISTS.

RELOCATED SPECIFICATIONS

None

No Significant Hazards Consideration

3.2.K and 4.2.K

Degraded Grid Protective System

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.

"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.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATION OF INSTRUMENTATION ONLY REQUIREMENTS
("LC.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 instrumentation requirements from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents. These requirements are not considered in the safety analysis. 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 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 requirements 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 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 assumption. In addition, the requirements 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 requirements 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 requirements 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 instrumentation requirements ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.K/4.2.K – DEGRADED GRID PROTECTIVE SYSTEM 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 relax the actions when one or more channels of Degraded Bus Voltage - Time Delay Trip Function channels are inoperable by providing one additional hour to restore the inoperable channels. The Degraded Bus Voltage – Time Delay instrumentation is not assumed to be an initiator of any analyzed event. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Continued operation with inoperable Degraded Bus Voltage – Time Delay Trip Function instrumentation channels will still 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 additional 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 a Degraded Bus Voltage – Time Delay 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 Degraded Bus Voltage – Time Delay Trip Function 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.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS 3.2.K/4.2.K – DEGRADED GRID PROTECTIVE SYSTEM 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 provides an additional action to declare the associated diesel generator inoperable and take the appropriate actions when one or more Degraded Bus Voltage – Voltage channels are inoperable, in lieu of a plant shutdown. The Degraded Bus Voltage – Voltage instrumentation is not assumed to be an initiator of any analyzed event. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Continued operation with inoperable Degraded Bus Voltage – Voltage Trip Function instrumentation channels will still be limited in accordance with Technical Specifications. The role of this instrumentation, in conjunction with an Emergency Core Cooling System initiation signal, is to provide a start signal for the diesel generators (i.e., it supports diesel generator operability). The proposed change to the actions will not allow continuous operation such that a single failure will preclude diesel generator initiation from mitigating event consequences. 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 a Degraded Bus Voltage – Voltage 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 provides an additional action to declare the associated diesel generator inoperable and take the appropriate actions when one or more Degraded Bus Voltage – Voltage channels are inoperable, in lieu of a plant shutdown. 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.K and 4.2.K

Degraded Grid Protective System

3.2.K/4.2.K REFERENCES
Degraded Grid Protective System

1. UFSAR, Section 8.5.3.
2. UFSAR, Section 6.5
3. UFSAR Chapter 14.

Proposed

Technical Specifications

3.2.L and 4.2.L

Reactor Core Isolation Cooling (RCIC) System Actuation

3.2 LIMITING CONDITIONS FOR OPERATION

L. Reactor Core Isolation Cooling (RCIC) System Actuation

The RCIC System instrumentation for each Trip Function in Table 3.2.9 shall be operable in accordance with Table 3.2.9.

4.2 SURVEILLANCE REQUIREMENTS

L. Reactor Core Isolation Cooling (RCIC) System Actuation

1. The RCIC System instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.9.

   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 as follows: (a) for up to 6 hours for Trip Function 3; and (b) for up to 6 hours for Trip Functions 1 and 2 provided the associated Trip Function maintains RCIC initiation capability.

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

Table 3.2.9 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Low-Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY(1), HOT SHUTDOWN(1), REFUEL(1)	2	Note 1	≥ 82.5 inches
2. Low Condensate Storage Tank Water Level	RUN, STARTUP/HOT STANDBY(1), HOT SHUTDOWN(1), REFUEL(1)	2	Note 2	≥ 3.81%(2)
3. High Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY(1), HOT SHUTDOWN(1), REFUEL(1)	2	Note 3	≤ 177.0 inches

(1) With reactor steam pressure > 150 psig.

(2) Percent of instrument span.

Table 3.2.9 ACTION Notes

1. With one or more RCIC System instrumentation Trip Function 1 channels inoperable:

   a. Declare the RCIC System inoperable within 1 hour from discovery of loss of RCIC initiation capability; and
   b. Place inoperable channel in trip within 24 hours.

   If any applicable Action and associated completion time of Note 1.a or 1.b is not met, immediately declare the RCIC System inoperable.

2. With one or more RCIC System instrumentation Trip Function 2 channels inoperable:

   a. Declare the RCIC System inoperable within 1 hour from discovery of loss of RCIC initiation capability when RCIC System suction is aligned to the Condensate Storage Tank; and
   b. Place inoperable channel in trip or align RCIC System suction to the suppression pool within 24 hours.

   If any applicable Action and associated completion time of Note 2.a or 2.b is not met, immediately declare the RCIC System inoperable.

3. With one or more RCIC System instrumentation Trip Function 3 channels inoperable:

   a. Restore inoperable channel to operable status within 24 hours.

   If the Action and associated completion time of Note 3.a is not met, immediately declare the RCIC System inoperable.

Table 4.2.9 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation
Tests and Frequencies

TRIP FUNCTION	CHECK	FUNCTIONAL TEST	CALIBRATION
1. Low-Low Reactor Vessel Water Level	Once/Day	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
2. Low Condensate Storage Tank Water Level	NA	Every 3 Months	Every 3 Months(1), Once/Operating Cycle
3. High Reactor Vessel Water Level	NA	Every 3 Months	Every 3 Months(1), Once/Operating Cycle

(1) Trip unit calibration only.

Proposed

Bases

3.2.L and 4.2.L

Reactor Core Isolation Cooling (RCIC) System Actuation

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

BACKGROUND

The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is insufficient or unavailable, such that RCIC System initiation occurs and maintains sufficient reactor water level such that initiation of the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. A more complete discussion of the RCIC System is provided in UFSAR, Section 4.7 (Ref. 1).

RCIC System automatic initiation occurs for conditions of Low - Low Reactor Vessel Water Level. The variable is monitored by four transmitters that are connected to four trip units. The Low - Low Reactor Vessel Water Level Trip Function is a single trip system with two trip system logics. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic arrangement.

The RCIC test line isolation valve is closed on a RCIC initiation signal to allow full system flow.

The RCIC System also monitors the water level in the condensate storage tank (CST) since this is the initial source of water for RCIC operation. Reactor grade water in the CST is the normal source. Upon receipt of a RCIC initiation signal, the CST suction valve is automatically signaled to open. If the water level in the CST falls below a preselected level, the RCIC suppression pool suction valves automatically open. When the suppression pool suction valves are both fully open, the RCIC CST suction valve automatically closes. Two level transmitters are used to detect low water level in the CST. Either transmitter can cause the suppression pool suction valves to open and the CST suction valve to close (one trip system arranged in a one-out-of-two logic).

The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the high water level trip (one trip system arranged in a two-out-of-two logic), at which time the RCIC steam admission valve closes. The RCIC System automatically restarts if a Low - Low Reactor Vessel Water Level signal is subsequently received.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The function of the RCIC System to provide makeup coolant to the reactor is used to respond to transient events. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system, and therefore its instrumentation, meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The operability of the RCIC System Instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions. 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 the actual trip setpoints within the calculational as-found tolerances provides reasonable

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.9. 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 individual Trip Functions are required to be operable in the RUN Mode and in the STARTUP/HOT STANDBY, HOT SHUTDOWN, and REFUEL Modes with reactor steam pressure > 150 psig since this is when RCIC is required to be operable.

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

1. Low - Low Reactor Vessel Water Level

Low reactor pressure vessel (RPV) water level indicates that normal feedwater flow is insufficient to maintain reactor vessel water level and that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the RCIC System is initiated on a Low - Low Reactor Vessel Water Level signal to assist in maintaining water level above the top of the enriched fuel.

Low - Low 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.

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). The Trip Setting is referenced from the top of enriched fuel.

Four channels of Low - Low Reactor Vessel Water Level Trip Function are available and are required to be operable when RCIC is required to be operable to ensure that no single instrument failure can preclude RCIC initiation.

2. Low Condensate Storage Tank Water Level

Low water level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valve between the RCIC pump and the CST is open and, upon receiving a RCIC initiation signal, water for RCIC injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, the RCIC suppression pool suction valves automatically open. When the suppression pool suction valves are both fully open, the RCIC CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump.

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Two level transmitters are used to detect low water level in the CST. The Low Condensate Storage Tank Water Level Trip Function Trip Setting is set high enough to ensure adequate pump suction head while water is being taken from the CST. The trip setting is presented in terms of percent instrument span.

Two channels of Low Condensate Storage Tank Water Level Trip Function are available and are required to be operable when RCIC is required to be operable to ensure that no single instrument failure can preclude RCIC swap to the suppression pool source.

3. High Reactor Vessel Water Level

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the high water level signal is used to close the RCIC steam admission valve to prevent overflow into the main steam lines (MSLs).

High Reactor Vessel Water Level signals for RCIC are initiated from two level transmitters, which 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.

The High Reactor Vessel Water Level Trip Setting is high enough to preclude closing the RCIC steam admission valve during normal operation, yet low enough to trip the RCIC System to prevent reactor vessel overfill. The Trip Setting is referenced from the top of enriched fuel.

Two channels of High Reactor Vessel Water Level Trip Function are available and are required to be operable when RCIC is required to be operable.

ACTIONS

Table 3.2.9 ACTION Note 1

Table 3.2.9 ACTION Note 1.a is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels of Trip Function 1 result in a complete loss of automatic initiation capability for the RCIC System. In this case, automatic initiation capability is lost if two Trip Function 1 channels in the same trip system logic are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Table 3.2.9 ACTION Note 1.b is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability. The completion time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This completion time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Table 3.2.9 ACTION Note 1.a, the completion time only begins upon discovery that the RCIC System cannot be automatically

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

ACTIONS (continued)

initiated due to two inoperable, untripped Low - Low Reactor Vessel Water Level channels in the same trip system logic. The 1 hour completion time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 2) to permit restoration of any inoperable channel to operable status. 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 3.2.9 ACTION Note 1.b. 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 any required Action and associated completion time of Table 3.2.9 ACTION Note 1.a or 1.b not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

Table 3.2.9 ACTION Note 2

Table 3.2.9 ACTION 2.a is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels of Trip Function 2 result in automatic RCIC initiation (i.e., suction swap) capability being lost. In this case, automatic RCIC suction swap capability is lost if two Trip Function 2 channels are inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Table 3.2.9 ACTION Note 2.b is not appropriate, and the RCIC System must be declared inoperable within 1 hour from discovery of loss of RCIC initiation capability when the RCIC System suction is aligned to the CST. Table 3.2.9 ACTION Note 1.a is only applicable if the RCIC System suction is not aligned to the suppression pool since, if aligned, the Trip Function is already performed. The completion time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This completion time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Table 3.2.9 ACTION Note 2.a, the completion time only begins upon discovery that the RCIC System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in Trip Function 2. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 2) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to operable

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

ACTIONS (continued)

status within the allowable out of service time, the channel must be placed in the tripped condition per Table 3.2.9 ACTION Note 2.b, which performs the intended function of the channel (shifting the suction source to the suppression pool). Alternatively, Table 3.2.9 ACTION Note 2.b allows the manual alignment of the RCIC System suction to the suppression pool, which also performs the intended function. If either action of Table 3.2.9 ACTION Note 2.b is performed, measures should be taken to ensure that the RCIC System piping remains filled with water.

With any required Action and associated completion time of Table 3.2.9 ACTION Note 2.a or 2.b not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

Table 3.2.9 ACTION Note 3

A risk based analysis was performed and determined that an allowable out of service time of 24 hours (Ref. 2) is acceptable to permit restoration of any inoperable Trip Function 3 channel to operable status (Table 3.2.9 ACTION Note 3.a). A required Action (similar to Table 3.2.9 ACTION Note 1.a) limiting the allowable out of service time, if a loss of automatic RCIC initiation capability (i.e., loss of high water level trip capability) exists, is not required. Table 3.2.9 ACTION Note 3 applies to the High Reactor Vessel Water Level Trip Function whose logic is arranged such that any inoperable channel will result in a loss of automatic RCIC initiation capability. As stated above, this loss of automatic RCIC initiation capability was analyzed and determined to be acceptable. One inoperable channel may result in a loss of high water level trip capability but will not prevent RCIC System automatic start capability. However, the Required Action does not allow placing a channel in trip since this action would not necessarily result in a safe state for the channel in all events (a failure of the remaining channel could prevent a RCIC System start).

With any required Action and associated completion time of Table 3.2.9 ACTION Note 3.a not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.2.L.1

As indicated in Surveillance Requirement 4.2.L.1, RCIC System instrumentation shall be checked, functionally tested and calibrated as indicated in Table 4.2.9. Table 4.2.9 identifies, for each Trip Function, the applicable Surveillance Requirements.

Surveillance Requirement 4.2.L.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

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (continued)

as follows: (a) for up to 6 hours for Trip Function 3; and (b) for up to 6 hours for Trip Functions 1 and 2, provided the associated Trip Function maintains RCIC initiation 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. 2) 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 RCIC System will initiate when necessary.

Surveillance Requirement 4.2.L.2

The Logic System Functional Test demonstrates the operability of the required initiation logic for a specific channel and includes simulated automatic actuation of the channel. The system functional testing performed in Surveillance Requirement 4.5.G.1 overlaps this Surveillance to provide complete testing of the 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 4.2.9, Check

Performance of an Instrument Check once per day, for Trip Function 1, 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.

BASES: 3.2.L/4.2.L REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (continued)

Table 4.2.9, Functional Test

For Trip Functions 1, 2 and 3, 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. For Trip Functions 1, 2 and 3, the Frequency of "Every 3 Months" is based on the reliability analysis of Reference 2.

Table 4.2.9, Calibration

For Trip Functions 1, 2, and 3, 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 3, 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 2 and the time interval assumption for trip unit calibration used in the associated setpoint calculation.

REFERENCES

1. UFSAR, Section 4.7.

2. GENE-770-06-2P-A, Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

Current

Technical Specifications

Markups

3.2.L and 4.2.L

Reactor Core Isolation Cooling (RCIC) System Actuation

A.1

3.2 LIMITING CONDITIONS FOR OPERATION

sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.

A.2

I. Recirculation Pump Trip Instrumentation

During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operable in accordance with Table 3.2.1.

J. Deleted

K. Degraded Grid Protective System

During reactor power operation, the emergency bus undervoltage instrumentation shall be operable in accordance with Table 3.2.8.

(RCIC)

L. Reactor Core Isolation Cooling System Actuation

A.3

When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G, the instrumentation which initiates actuation of this system shall be operable in accordance with Table 3.2.9.

RCIC SYSTEM

FOR EACH TRIP FUNCTION IN TABLE 3.2.9

4.2 SURVEILLANCE REQUIREMENTS

(MOVE TO SEPARATE PAGE)

I. Recirculation Pump Trip Instrumentation

The Recirculation Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.

J. Deleted

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.

L. Reactor Core Isolation Cooling System Actuation

RCIC SYSTEM

Instrumentation and Logic Systems shall be functionally tested and calibrated as indicated in Table 4.2.9.

A.4

CHECKED,

A.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 AS FOLLOWS: (a) FOR UP TO 6 HOURS FOR TRIP FUNCTION 3; AND (b) FOR UP TO 6 HOURS FOR TRIP FUNCTIONS 1 AND 2 PROVIDED THE ASSOCIATED TRIP FUNCTION MAINTAINS RCIC INITIATION CAPABILITY.

A.6

A.1

REQUIRED CHANNELS ARE INOPERABLE

TABLE 3.2.9

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

REQUIRED

	Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required ACTIONS When Minimum Conditions For Operation Are Not Satisfied
1	2 (Notes 1, 5) A.6	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(M)) LA.1	≥82.5" Above Top of Enriched Fuel LA.3	Note 1
2	2 (Notes 2, 5)	Low Condensate Storage Tank Water Level (LT-107-12A/B(M)) LA.1	≥3% (a) 3.81% M.2	Note 2
3	2 (Notes 3, 6) LA.2	High Reactor Vessel Water Level (LT-2-3-72C/D(S2)) LA.1	≤177" Above Top of Enriched Fuel LA.3	Note 3
	1	Trip System Logic	--	Note 4 L.1

(a) PERCENT OF INSTRUMENT SPAN.

A.1

TABLE 3.2.9 ACTION NOTES

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

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

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

LA.2

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

L.1

(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 6 hours provided the associated Trip Function maintains RCIC initiation capability.

A.6

(6) 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.

Table 3.2.9 ACTION Note 1

(7) With one or more channels inoperable for RCIC:

a (A) Within one hour from discovery of loss of system initiation capability, declare the RCIC system inoperable, and

b (B) Within 24 hours, place channel in trip.

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

Table 3.2.9 ACTION Note 2

(8) With one or more channels inoperable for RCIC:

a (A) Within one hour from discovery of loss of system initiation capability while suction is aligned to the CST, declare the RCIC system inoperable, and

b (B) Within 24 hours, place channel in trip or align suction for the RCIC system to the suppression pool.

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

Table 3.2.9 ACTION Note 3

(9) With one or more channels inoperable for RCIC:

(A) Within 24 hours, restore channel to operable status.

(B) If required action and associated completion time of action A is not met, immediately declare the RCIC system inoperable.

A.1

TABLE 4.2.9

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

REACTOR CORE ISOLATION COOLING SYSTEM ~~ACTUATION~~ INSTRUMENTATION

	Trip Function	Functional Test(~~s~~) A.7	Calibration(~~s~~) A.7	~~Instrument~~ Check
1.	Low-Low Reactor Vessel Water Level	Every Three Months	Once/Operating Cycle	Once each day
2.	Low Condensate Storage Tank Water Level	Every Three Months	Once/Operating Cycle	--
3.	High Reactor Vessel Water Level	Every Three Months	Once/Operating Cycle	--
SR4.2.1.2	Trip System Logic	Once/Operating Cycle	~~Once/Operating Cycle (Note 3)~~ A.8	-- A.4

Every 3 Months,(1) M.1

(1) Trip unit calibration only. M.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. A.9

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system. A.8

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.

8. Functional tests and calibrations are not required when systems are not required to be operable. A.7

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.L and 4.2.L

Reactor Core Isolation Cooling (RCIC) System Actuation

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.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.

A.3 CTS 3.2.L specifies an Applicability for Reactor Core Isolation Cooling (RCIC) System instrumentation of "When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G." Specification 3.5.G includes the requirements for the RCIC System. This change provides an explicit Applicability, in proposed Table 3.2.9 for each RCIC System instrumentation Trip Function. The specified Applicabilities, in proposed Table 3.2.9, are consistent with the Modes and conditions when the RCIC System are required to be operable by Specification 3.5.G. 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.4 CTS 4.2.L specifies that instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.9. In proposed Surveillance Requirement (SR) 4.2.L.1, the reference to "and logic system," is deleted since associated logic systems are considered part of the RCIC System instrumentation Trip Functions in both proposed and CTS Tables 3.2.9 and 4.2.9. It is not necessary to explicitly identify logic systems in CTS 4.2.L, since proposed SR 4.2.L.2 (CTS Table 4.2.9 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 RCIC System instrumentation Trip Function). Therefore, this change is considered administrative.

A.5 CTS 4.2.L includes reference to CTS Table 4.2.9 for functional test and calibration requirements for RCIC System instrumentation. CTS 4.2.I is revised, in proposed SR 4.2.L.1, to also include reference to check requirements consistent with CTS Table 4.2.9. This change is a presentation preference and does not alter the current requirements to periodically perform checks of certain RCIC System instrument Trip Functions. Therefore, this change is considered administrative in nature.

ADMINISTRATIVE (continued)

A.6 CTS Table 3.2.9, Notes 5 and 6, 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.L.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.7 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.8 For the Trip System Logic associated with the RCIC System instrumentation, CTS Table 4.2.9 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 RCIC System instrumentation Trip Functions of CTS Table 4.2.9 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 4.2.9, the RCIC System instrumentation (proposed Trip Functions 1, 2, and 3) 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.9 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 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 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 Table 4.2 Notes 4, 12, and 13 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Table 4.2.9 does not include explicit requirements to calibrate trip units. Proposed Table 4.2.9 requires calibration of the trip units of the following Trip Functions every 3 months: Low-Low Reactor Vessel Water Level (proposed Table 4.2.9 Trip Function 1), Low Condensate Storage Tank Water Level (proposed Table 4.2.9 Trip Function 2), and High Reactor Vessel Water Level (proposed Table 4.2.9 Trip Function 3). The trip units of these Trip Functions are currently required by CTS Table 4.2.9 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.2 CTS Table 3.2.9 specifies for the Low Condensate Storage Tank Water Level Trip Function that the Trip Setting be ≥ 3%. The function of the Low Condensate Storage Tank Water Level is to provide an automatic transfer of the RCIC suction source from the condensate storage tank to the suppression pool when the level in the condensate storage tank is no longer sufficient to support adequate RCIC pump suction head. The CTS Trip Setting has been determined to be insufficient to ensure that transfer of the RCIC System suction from the condensate storage tank to the suppression pool occurs prior to potential vortex formation at the RCIC suction inlet in the condensate storage tank. Therefore, in proposed Table 3.2.9, the Trip Setting for the Low Condensate Storage Tank Water Level Trip Function (Trip Function 2) has been increased to ≥ 3.81% to account for the additional water level needed to preclude the potential for vortex formation. This minimum level corresponds to the Process Limit used in the associated setpoint calculation. To account for instrument uncertainties, the instrument setpoint and as-found tolerance (i.e., instrument operability limit) were developed using the Vermont Yankee Instrument Uncertainty and Setpoints Design Guide. Footnote (2) in proposed Table 3.2.9 clarifies that the trip setting is specified in terms of percent instrument span. The instrument setpoint and as-found tolerance are located in plant procedures. This change represents an additional restriction on plant operation necessary to ensure that RCIC System operability is maintained when aligned to the condensate storage tank and that RCIC pump suction transfer to the suppression pool occurs prior to the vortex formation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 3.2.9 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.L and Table 3.2.9 require the RCIC System Instrumentation Trip Functions to be operable. In addition, the proposed Surveillance Requirements in Table 4.2.9 ensure the required instruments are properly tested. These requirements are adequate for ensuring each of the required RCIC System 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

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.1 (continued) 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 3.2.9 Notes 1, 2, and 3 contain design details of the RCIC System instrumentation (i.e., one trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic, one trip system with initiating instrumentation arranged in a one-out-of-two logic, and one trip system arranged in a two-out-of-two logic). These details are not necessary to ensure the operability of RCIC System instrumentation. Therefore, the information in these notes is to be relocated to Specification 3.2.L Bases and reference to this information is deleted from VYNPS TS. The requirements of Specification 3.2.L and the associated Surveillance Requirements for the RCIC System 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.3 The Trip Settings associated with reactor vessel water level trip functions (proposed Table 3.2.9 Trip Functions 1 and 3) are currently referenced to "Above 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 RCIC System instrumentation. Operability requirements are adequately addressed in proposed Specification 3.2.L, Table 3.2.9 and the specified Trip Settings. 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.

"Specific"

L.1 CTS Table 3.2.9 includes requirements for Trip System Logics associated with the RCIC System instrumentation Trip Functions. These Trip Systems Logics are considered part of the RCIC System 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 3.2.9 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 3.2.9 Note 4) will now be governed by the actions for the individual proposed Table 3.2.9 RCIC System instrumentation Trip Functions. These proposed Table 3.2.9 Action Notes are less restrictive than the CTS Table 3.2.9 Note 4 actions. However, the proposed actions will ensure, in the event of inoperabilities, that consistent actions are applied to both RCIC System 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 3.2.9 Action Notes will limit operation to within the bounds of the applicable analysis, i.e., GENE-770-06-2-A, "Addendum to Bases for Changes to

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.1 (continued) Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992. Application of these analyses to the VYNPS RCIC System 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.L and 4.2.L

Reactor Core Isolation Cooling (RCIC) System 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 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.

"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.L/4.2.L – REACTOR CORE ISOLATION COOLING SYSTEM 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 relax the actions when one or more channels of Reactor Core Isolation Cooling (RCIC) System Trip Function instrumentation are inoperable due to inoperable Trip System Logic. The RCIC System 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 RCIC System Trip Function instrumentation channels will still 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 RCIC System 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 RCIC System 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.L and 4.2.L

Reactor Core Isolation Cooling (RCIC) System Actuation

3.2.L/4.2.L REFERENCES
Reactor Core Isolation Cooling (RCIC) System Actuation

1. UFSAR, Section 4.7.

2. GENE-770-06-2P-A, Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

Proposed

Technical Specifications Bases

3.1/4.1 and 3.2/4.2

Reactor Protection System Bases
and
Protective Instrument System Bases

BASES:

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings of Section 2.1 are discussed in the Bases for Specifications 3.1.A, 3.2.A and 3.2.B.

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Current Technical Specifications Bases

Markups

3.1/4.1 and 3.2/4.2

Reactor Protection System Bases and Protective Instrument System Bases

A.1

BASES:

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings are discussed in ~~the following paragraphs.~~ OF SECTION 2.1 THE BASES FOR SPECIFICATIONS 3.1.A, 3.2.A, AND 3.2.B.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses are performed to demonstrate that the APRM flux scram over the range of settings from a maximum of 120% to the minimum flow biased setpoint of 54% provide protection from the fuel safety limit for all abnormal operational transients including those that may result in a thermal hydraulic instability.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. If the scram requires a change due to an abnormal peaking condition, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1.A.1.a, thus assuring a reactor scram at lower than design overpower conditions. For single recirculation loop operation, the APRM flux scram trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR defined in the Core Operating Limits Report.

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BASES: 2.1 (Cont'd)

Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psig.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

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BASES: 2.1 (Cont'd)

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

B. Deleted

C. Reactor Low Water Level Scram

The reactor low water level scram is set at a point which will prevent reactor operation with the steam separators uncovered, thus limiting carry-under to the recirculation loops. In addition, the safety limit is based on a water level below the scram point and therefore this setting is provided.

D. Reactor Low Water Level ECCS Initiation Trip Point

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to well below the clad melting temperature, and to limit clad metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the ECCS initiation setpoint would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

A.1

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BASES: 2.1 (Cont'd)

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram signal may be bypassed at ≤30% of reactor Rated Thermal Power.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists. This scram signal may be bypassed at ≤30% of reactor Rated Thermal Power.

G. Main Steam Line Isolation Valve Closure Scram

The isolation scram anticipates the pressure and flux transients which occur during an isolation event and the loss of inventory during a pipe break. This action minimizes the effect of this event on the fuel and pressure vessel.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

BASES:

3.1 Reactor Protection System

The reactor protection system automatically initiates a reactor scram to:

1. preserve the integrity of the fuel barrier;
2. preserve the integrity of the primary system barrier; and
3. minimize the energy which must be absorbed, and prevent criticality following a loss of coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance, testing, or calibration. The basis for the allowable out-of-service times is provided in GE Topical Report NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," March 1988.

The reactor protection system is of the dual channel type. The system is made up of two independent logic channels, each having three subsystems of tripping devices. One of the three subsystems has inputs from the manual scram push buttons and the reactor mode switch. Each of the two remaining subsystems has an input from at least one independent sensor monitoring each of the critical parameters. The outputs of these subsystems are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subsystems will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both logic channels is required to produce a reactor scram.

The required conditions when the minimum instrument logic conditions are not met are chosen so as to bring station operation promptly to such a condition that the particular protection instrument is not required; or the station is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions.

When the minimum requirements for the number of operable or operating trip system and instrumentation channels are satisfied, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

Three APRM instrument channels are provided for each protection trip system to provide for high neutron flux protection. APRM's A and E operate contacts in a trip subsystem, and APRM's C and E operate contacts in the other trip subsystem. APRM's B, D, and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration without changing the minimum number of channels required for inputs to each trip system. Additional IRM channels have also been provided to allow bypassing of one such channel. For a description of the Neutron Monitoring Systems, see FSAR Section 7.5.

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specification 2.1.

A.1

BASES: 3.1 (Cont'd)

Instrumentation is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The Control Rod Drive Scram System is designed so that all of the water that is discharged from the reactor by the scram can be accommodated in the discharge piping. This discharge piping is divided into two sections. One section services the control rod drives on the north side of the reactor, the other serves the control rod drives of the south side. A part of the piping in each section is an instrument volume which accommodates in excess of 21 gallons of water and is at the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level instrumentation has been provided for the instrument volume which scram the reactor when the volume of water reaches 21 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water, and precludes the situation in which a scram would be required but not be able to perform its function adequately. The present design of the Scram Discharge System is in concert with the BWR Owner's Group criteria, which have previously been endorsed by the NRC in their generic "Safety Evaluation Report (SER) for Scram Discharge Systems", dated December 1, 1980.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass.

Turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure scram signals may be bypassed at $\leq$30% of reactor Rated Thermal Power since, at low thermal power levels, the margins to fuel thermal-hydraulic limits and reactor primary coolant boundary pressure limits are large and an immediate scram is not necessary. This bypass function is normally accomplished automatically by pressure switches sensing turbine first stage pressure. The turbine first stage pressure setpoint controlling the bypass of the scram signals on TCV fast closure and TSV closure is derived from analysis of reactor pressurization transients. Certain operational factors, such as turbine bypass valves open, can influence the relationship between turbine first stage pressure and reactor Rated Thermal Power. However, above 30% of reactor Rated Thermal Power, these scram functions must be enabled.

BASES: 3.1 (Cont'd)

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10 percent closed from full open in 3-out-of-4 lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status.

The manual scram function is active in all modes, thus providing for manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against short reactor periods and, in conjunction with the reduced APRM system provides protection against excessive power levels in the startup and intermediate power ranges. A source range monitor (SRM) system is also provided to supply additional neutron level information during startup and can provide scram function with selected shorting links removed during refueling. Thus, the IRM and the reduced APRM are normally required in the startup mode and may be required in the refuel mode. During some refueling activities which require the mode switch in startup; it is allowable to disconnect the LPRMs to protect them from damage during under vessel work. In lieu of the protection provided by the reduced APRM scram, both the IRM scram and the SRM scram in noncoincidence are used to provide neutron monitoring protection against excessive power levels. In the power range, the normal APRM system provides required protection. Thus, the IRM system and 15% APRM scram are not required in the run mode.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criteria. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system.

Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped system until the surveillance testing deadline.

BASES: 3.1 (Cont'd)

The requirement to have all scram functions, except those listed in Table 3.1.1, operable in the "Refuel" mode is to assure that shifting to this mode during reactor operation does not diminish the need for the reactor protection system.

The ability to bypass one instrument channel when necessary to complete surveillance testing will preclude continued operation with scram functions which may be either unable to meet the single failure criteria or completely inoperable. It also eliminates the need for an unnecessary shutdown if the remaining channels and subsystems are found to be operable. The conditions under which the bypass is permitted require an immediate determination that the particular function is operable. However, during the time a bypass is applied, the function will not meet the single failure criteria; therefore, it is prudent to limit the time the bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed.

Sluggish indicator response during the perturbation test will be indicative of a plugged instrument line or closed instrument valves. This test assures the operability of the reactor pressure sensors as well as the reactor level sensors since both parameters are monitored through the same instrument lines.

The independence of the safety system circuitry is determined by operation of the scram test switch. Operation of this switch during the refueling outage and following maintenance on these circuits will assure their continued independence.

A.1

BASES: 4.1 REACTOR PROTECTION SYSTEM

A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B and C. Sensors that make up Group A are the on-off type and will be tested and calibrated at the indicated intervals.

Group B devices utilize an analog sensor followed by an amplifier and bistable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bistable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during start-up and inactive during full-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

The basis for a three-month functional test interval for group (A) and (B) sensors is provided in NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection Systems," March 1988.

SRM/IRM/APRM overlap Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between the RUN and STARTUP/HOT STANDBY Modes can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

As noted, IRM/APRM overlap is only required to be met during entry into STARTUP/HOT STANDBY Mode from the Run Mode. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in the STARTUP/HOT STANDBY Mode).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current condition should be declared inoperable.

A.1

BASES: 4.1 (Cont'd)

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 2,000 megawatt-days per short ton (MWD/T) frequency is based on operating experience with LPRM sensitivity changes, and that the resulting nodal power uncertainty, combined with other identified uncertainties, remains less than the total uncertainty (i.e., 8.7%) allowed by the GETAB safety limit analysis.

B. The ratio of MFLPD to FRP shall be checked once per day when operating at ≥25% Rated Thermal Power to determine if the APRM gains require adjustment. Because few control rod movements or power changes occur, checking these parameters daily is adequate. 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.

A.1

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated.

A.1

BASES: 3.2 (Cont'd)

For the complete circumferential break of 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Certain isolation valves including the TIP blocking valves, CAD inlet and outlet, drywell vent, purge and sump valves are isolated on high drywell pressure. However, since high drywell pressure could occur as the result of non-safety-related causes, such as not venting the drywell during startup, complete system isolation is not desirable for these conditions and only certain valves are required to close. The water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of certain primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR100 are exceeded.

Isolation of the condenser mechanical vacuum pump (MVP) is assumed in the safety analysis for the control rod drop accident (CRDA). The MVP isolation instrumentation initiates closure of the MVP suction isolation valve following events in which main steam line radiation monitors exceed a predetermined value. A High Main Steam Line Radiation Monitor trip setting for MVP isolation of ≤ 3 times background at rated thermal power (RTP) is as low as practicable without consideration of spurious trips from nitrogen-16 spikes, instrument instabilities and other operational occurrences. Isolating the condenser MVP limits the release of fission products in the event of a CRDA.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the

A.1

BASES: 3.2 (Cont'd)

control and/or bypass valves to open, resulting in a rapid depressurization and cooldown of the reactor vessel. The 800 psig trip setpoint limits the depressurization such that no excessive vessel thermal stress occurs as a result of a pressure regulator malfunction. This setpoint was selected far enough below normal main steam line pressures to avoid spurious primary containment isolations.

Low condenser vacuum has been added as a trip of the Group 1 isolation valves to prevent release of radioactive gases from the primary coolant through condenser. The setpoint of 12 inches of mercury absolute was selected to provide sufficient margin to assure retention capability in the condenser when gas flow is stopped and sufficient margin below normal operating values.

The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. The HPCI and RCIC steam supply pressure instrumentation is provided to isolate the systems when pressure may be too low to continue operation. 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 if not properly tested. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves, i.e., Group 6 valves. A time delay has been incorporated into the RCIC steam flow trip logic to prevent the system from inadvertently isolating due to pressure spikes which may occur on startup. The trip settings are such that core uncovering is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual channel system. Permanently installed circuits and equipment may be used to trip instrument channels. In the nonfail safe systems which require energizing the circuitry, tripping an instrument channel may take the form of providing the required relay function by use of permanently installed circuits. This is accomplished in some cases by closing logic circuits with the aid of the permanently installed test jacks or other circuitry which would be installed for this purpose.

The Rod Block Monitor (RBM) control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease below the fuel cladding integrity safety limit. The RBM is credited in the Continuous Rod Withdrawal During Power Range Operation transient for preventing excessive control rod withdrawal before the fuel cladding integrity safety limit (MCPR) or the fuel rod mechanical overpower limits are exceeded. The RBM upper limit is clamped to provide protection at greater than 100% rated core flow. The clamped value is cycle specific; therefore, it is located in the Core Operating Limits Report.

For single recirculation loop operation, the RBM trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

During hot shutdown, cold shutdown, and refueling when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical with sufficient shutdown margin; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch-Shutdown Position control rod withdrawal block, required to be operable with the mode switch in the shutdown position, ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis. Two channels are required to be

A.1

BASES: 3.2 (Cont'd)

operable to ensure that no single channel failure will preclude a rod block when required. There is no trip setting for this function since the channels are mechanically actuated based solely on reactor mode switch position. During refueling with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock provides the required control rod withdrawal blocks.

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since, for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. For a break or other event occurring outside the drywell, the Automatic Depressurization System is initiated on low-low reactor water level only after a time delay. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the Specification are adequate to ensure the above criteria are met. The Specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

The ADS is provided with inhibit switches to manually prevent automatic initiation during events where actuation would be undesirable, such as certain ATWS events. The system is also provided with an Appendix R inhibit switch to prevent inadvertent actuation of ADS during a fire which requires evacuation of the Control Room.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. Any one upscale trip or two downscale trips of either set of monitors will cause the desired action. Trip settings for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leave the Reactor Building via the normal ventilation stack but that all activity is processed by the standby gas treatment system. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation at a radiation level equivalent to the maximum site boundary dose rate of 500 mrem/year as set forth in the Offsite Dose Calculation Manual. The monitoring system in the plant stack represents a backup to this system to limit gross radioactivity releases to the environs.

The purpose of isolating the mechanical vacuum pump line is to limit release of radioactivity from the main condenser. During an accident, fission products would be transported from the reactor through the main steam line to the main condenser. The fission product radioactivity would be sensed by the main steam line radiation monitors which initiate isolation.

BASES: 3.2 (Cont'd)

Specification 3.2.G requires that the post-accident monitoring (PAM) instrumentation of Table 3.2.6 be operable during reactor power operation. PAM instrumentation is not required to be operable during shutdown and refueling conditions when the likelihood of an event that would require PAM instrumentation is extremely low. The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents. The operability of the PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

In most cases, Table 3.2.6 requires a minimum of two operable channels to ensure that the operators are provided the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following an accident. For the majority of parameters monitored, when one of the required channels is inoperable, the required inoperable channel must be restored to operable status within 30 days. The 30-day completion time is based on operating experience and takes into account the remaining operable channel (or in the case of a parameter that has only one required channel, an alternate means to monitor the parameter), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

If a PAM instrument channel has not been restored to an operable status within the specified interval, the required action is to prepare a written report to be submitted to the NRC within the following 14 days. When a special written report is required in accordance with the provisions of Table 3.2.6, the report will outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation to an operable status. This action is appropriate in lieu of a shutdown requirement, since alternative actions are identified before loss of functional capability, and given the likelihood of plant conditions that would require information provided by this instrumentation.

For the majority of PAM instrumentation, when two required channels are inoperable (or in the case of a parameter that is monitored by only one channel, the channel and an alternate means are inoperable), one channel (or the required alternate means) should be restored to an operable status within seven days. The completion time of seven days is based on the relatively low probability of an event requiring PAM instrumentation and the normal availability of alternate means to obtain the required information. Where specified, continuous operation with two required channels inoperable (or one channel and the required alternate means inoperable) is not acceptable after seven days. Therefore, restoration of one inoperable channel limits the risk that the PAM function will be in a degraded condition should an accident occur.

BASES: 3.2 (Cont'd)

For the majority of PAM instrumentation in Table 3.2.6, if two of the required channels (one required channel per valve and alternate means for safety valve position indication) remain inoperable beyond the allowed interval, actions must be taken to place the reactor in a mode or condition in which the limiting condition for operation does not apply. To achieve this status, the reactor must be brought to at least hot shutdown within 12 hours. The allowed completion time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. It is not necessary to bring the reactor to cold shutdown since plant conditions during hot shutdown are such that the likelihood of an accident that would require PAM instrumentation is extremely low.

The Degraded Grid Protective System has been installed to assure that safety-related electrical equipment will not be subjected to sustained degraded voltage. This system incorporates voltage relays on 4160 Volt Emergency Buses 3 and 4 which are set to actuate at the minimum voltage required to prevent damage of safety-related equipment.

If Degraded Grid conditions exist for 10 seconds, either relay will actuate an alarm to alert operators of this condition. Based upon an assessment of these conditions the operator may choose to manually disconnect the off-site power. In addition, if an ESF signal is initiated in conjunction with low voltage below the relay setpoint for 10 seconds, the off-site power will be automatically disconnected.

The Reactor Core Isolation Cooling (RCIC) System provides makeup water to the reactor vessel during shutdown and isolation to supplement or replace the normal makeup sources without the use of the Emergency Core Cooling Systems. The RCIC System is initiated automatically upon receipt of a reactor vessel low-low water level signal. Reactor vessel high water level signal results in shutdown of the RCIC System. However, the system will restart on a subsequent reactor vessel low-low water level signal. The RCIC System is normally lined up to take suction from the condensate storage tank. Suction will automatically switch over from the condensate storage tank to the suppression pool on low condensate storage tank level.

Upon receipt of a LOCA initiation signal, if normal AC power is available, all RHR pumps and both Core Spray pumps start simultaneously with no intentional time delay. If normal AC power is not available, RHR pumps A and D start immediately on restoration of power, RHR pumps B and C start within 3 to 5 seconds of restoration of power and both Core Spray pumps start within 8 to 10 seconds of restoration of power. The purpose of these time delays is to stagger the start of the RHR and Core Spray pumps on the associated Division 1 and Division 2 Buses, thus limiting the starting transients on the 4.16 kV emergency buses. The time delay functions are only necessary when power is being supplied from the standby power sources (EDGs). The time delays remain in the pump start logic at all times as the time delay relay contact is in parallel with the Auxiliary Power Monitor relay contact. Either contact closure will initiate pump start. Thus, the time delays do not affect low pressure ECCS pump operation with normal AC power available. With normal AC power not available, the pump start relays which would have started the B and C RHR pumps and both Core Spray pumps are blocked by the Auxiliary Power Monitor contacts and the pump start time delay relays become the controlling devices.

A.1

BASES:

4.2 PROTECTIVE INSTRUMENTATION

The Protective Instrumentation Systems covered by this Specification are listed in Table 4.2. Most of these protective systems are composed of two or more independent and redundant subsystems which are combined in a dual-channel arrangement. Each of these subsystems contains an arrangement of electrical relays which operate to initiate the required system protective action.

The relays in a subsystem are actuated by a number of means, including manually-operated switches, process-operated switches (sensors), bistable devices operated by analog sensor signals, timers, limit switches, and other relays. In most cases, final subsystem relay actuation is obtained by satisfying the logic conditions established by a number of these relay contacts in a logic array. When a subsystem is actuated, the final subsystem relay(s) can operate protective equipment, such as valves and pumps, and can perform other protective actions, such as tripping the main turbine generator unit.

With the dual-channel arrangement of these subsystems, the single failure of a ready circuit can be tolerated because the redundant subsystem or system (in the case of high pressure coolant injection) will then initiate the necessary protective action. If a failure in one of these circuits occurs in such a way that an action is taken, the operator is immediately alerted to the failure. If the failure occurs and causes no action, it could then remain undetected, causing a loss of the redundancy in the dual-channel arrangement. Losses in redundancy of this nature are found by periodically testing the relay circuits and contacts in the subsystems to assure that they are operating properly.

The surveillance test interval for the instrumentation channel functional tests are once/three months for most instrumentation. The allowable out-of-service times and surveillance interval is based on the following NRC approved licensing topical reports:

1. GE Topical Report NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," March 1988.

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

3. GE Topical Report NEDC-30851P-A, Supplement 2 "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

4. GE Topical Report NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation, " July 1990.

5. GE Topical Report NEDC-30936P-A, Parts 1 and 2, "BWR Owners Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," December 1988.

6. GE Topical Report GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

7. GE Topical Report GENE-770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

BASES:

4.2 PROTECTIVE INSTRUMENTATION (Cont'd)

Since logic circuit tests result in the actuation of plant equipment, testing of this nature was done while the plant was shut down for refueling. In this way, the testing of equipment would not jeopardize plant operation.

This Specification is a periodic testing program which is based upon the overall testing of protective instrumentation systems, including logic circuits as well as sensor circuits. Table 4.2 outlines the test, calibration, and logic system functional test schedule for the protective instrumentation systems. The testing of a subsystem includes a functional test of each relay wherever practicable. The testing of each relay includes all circuitry necessary to make the relay operate, and also the proper functioning of the relay contacts. Testing of the automatic initiation inhibit switches verifies the proper operability of the switches and relay contacts. Functional testing of the inaccessible temperature switches associated with the isolation systems is accomplished remotely by application of a heat source to individual switches.

All subsystems are functionally tested, calibrated, and operated in their entirety.

A channel functional test is performed for the Reactor Mode Switch - Shutdown Position function to ensure that the entire channel will perform the intended function. The surveillance is only required to be performed once per operating cycle during refueling. The Refueling Outage frequeency is based on the need to perform this surveillance under the conditions that apply during a plant outage. Operating experience has shown that this surveillance frequency is adequate to ensure functional operability. Note 12 of Table 4.2.5 specifies that if the surveillance frequency of 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 the Shutdown position for the purpose of commencing a scheduled Refueling Outage. This allows entry into the Shutdown mode when the surveillance requirement is not met.

Safety Assessment

Discussion of Changes

3.1/4.1 and 3.2/4.2

Reactor Protection System Bases
and
Protective Instrument System Bases

SAFETY ASSESSMENT OF CHANGES
TS SECTION 3.1/4.1 – REACTOR PROTECTION SYSTEM BASES
TS SECTION 3.2/4.2 – PROTECTIVE INSTRUMENT SYSTEM BASES

ADMINISTRATIVE

A.1 The Bases of the current Technical Specifications for Sections 3.1/4.1 and 3.2/4.2 (pages 29 through 33a for Section 3.1/4.1 and pages 75 through 80a for Section 3.2/4.2) are completely replaced by revised Bases that reflect the format and applicable content of the proposed Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications in Sections 3.1/4.1 and 3.2/4.2, consistent with the Boiling Water Reactor Improved Standard Technical Specifications NUREG-1433, Rev. 2. The revised Bases are as shown in the proposed VYNPS Technical Specification Bases. The Bases changes are made for clarity purposes and conformance to the changes being made to the associated Technical Specifications. In addition, the information on Trip Settings provided in the Bases for current Technical Specifications (CTS) Section 2.1 (pages 14 through17) is superceded by the revised Bases for Technical Specification Section 3.1/4.1 and no longer necessary. Therefore, this information in the Bases for CTS Section 2.1 is deleted and replaced with a statement referring to the Bases for the applicable Technical Specifications (i.e., the bases for the individual trip settings of Section 2.1 are discussed in the Bases for Specifications 3.1.A, 3.2.A, and 3.2.B). The Bases do not establish actual requirements, and as such do not change technical requirements in the Technical Specifications. Therefore, the changes are administrative in nature and have no negative impact on plant safety.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

None

No Significant Hazards Consideration

3.1/4.1 and 3.2/4.2

Reactor Protection System Bases
and
Protective Instrument System Bases

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.

NO SIGNIFICANT HAZARDS CONSIDERATION
TS SECTION 3.1/4.1 – REACTOR PROTECTION SYSTEM BASES
TS SECTION 3.2/4/2 – PROTECTIVE INSTRUMENT SYSTEM BASES

There were no specific less restrictive changes identified for these Bases.