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United States Nuclear Regulatory Commission
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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
STANDBY LIQUID CONTROL SYSTEM
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

In accordance with the requirements of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby transmits a request for revision of the Technical Specifications (TS) for Hope Creek Generating Station. Pursuant to the requirements of 10 CFR 50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed change revises Technical Specification (TS) 3.1.5, "Standby Liquid Control (SLC) System" APPLICABILITY section to be consistent with NUREG –1433, "Standard Technical Specifications General Electric Plants, BWR/4." It makes corresponding changes to the SLC Initiation sections of Table 3.3.2-1, "Isolation Actuation Instrumentation," and Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements." Similar changes were approved for Nine Mile Point Nuclear Station, Unit 2, by letter dated September 30, 1993 (TAC No. M86186), Susquehanna Steam Electric Station, Units 1 and 2, by letter dated December 20, 1994 (TAC Nos. M89666 & M89667), Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, by letter dated March 6, 1998 (TAC Nos. M99907, M99908, M99943, and M99944), and Duane Arnold Energy Center by letter dated March 31, 1998 (TAC No. MA0780).

PSEG has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested change is provided in Attachment 1 to this letter. The marked-up Technical Specification pages affected by the proposed changes are provided in Attachment 2. The retyped Technical Specification pages are provided in Attachment 3. For your information, Attachment 4 provides the existing TS Bases pages marked-up to reflect the associated changes to the TS.

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PSEG requests approval of this proposed change by April 1, 2006, in order to support Hope Creek Refueling Outage RF13, with implementation within 60 days of receipt of the approved amendment.

If you have any questions concerning this request, please contact Michael Jesse at (856) 339-1280.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 10/11/05

George P. Barnes
George P. Barnes
Site Vice President – Hope Creek

Attachments (4)

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

CHANGE TO TECHNICAL SPECIFICATIONS STANDBY LIQUID CONTROL (SLC) SYSTEM

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CHANGES TO TECHNICAL SPECIFICATIONS

1. DESCRIPTION

This letter is a request to amend Operating License NPF-57 for the Hope Creek Generating Station. The proposed change revises Technical Specification (TS) 3.1.5, "Standby Liquid Control (SLC) System" APPLICABILITY section to be consistent with NUREG –1433, "Standard Technical Specifications General Electric Plants, BWR/4," by removing the requirement for SLC to be OPERABLE in OPERATIONAL CONDITION 5* (REFUELING), with any control rod withdrawn. It makes corresponding changes to the SLC Initiation sections of Table 3.3.2-1, "Isolation Actuation Instrumentation," and Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements."

2. PROPOSED CHANGE

The proposed amendment changes the operability requirement for the SLC System from OPERATIONAL CONDITIONS 1 (POWER OPERATION), 2 (STARTUP), and 5* (REFUELING), with any control rod withdrawn to OPERATIONAL CONDITIONS 1 (POWER OPERATION) and 2 (STARTUP).

The associated action for OPERATIONAL CONDITION 5* and the footnote "*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2" are being deleted.

In addition, corresponding changes are being made to line item 4.e, "SLCS Initiation," on Table 3.3.2-1 and Table 4.3.2.1-1 to remove the applicability in OPERATIONAL CONDITION 5#. An associated footnote (#) in Table 3.3.2-1 and Table 4.3.2.1-1 is also being removed. These changes are editorial in nature as there is no need for isolation actuation instrumentation in an OPERATIONAL CONDITION where the system is not required. The footnote (#) that is being removed refers back to the SLC System Technical Specification for applicability.

3. BACKGROUND

The purpose of the SLC System, which is a reactivity control system, is to provide a backup capability for bringing the reactor from full power to a cold, xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. The design basis of the SLC System provides a specified cold shutdown boron concentration in the reactor core. The SLC System was designed to inject the cold shutdown boron concentration in 90 to 120 minutes. The time requirement was selected to override the reactivity insertion rate due to cool down following the xenon poison peak. The SLC

System satisfies the requirements of 10 CFR 50.62(c)(4) for reduction of risk from anticipated transient without scram (ATWS) events.

4. TECHNICAL ANALYSIS

In order to reduce the risk from ATWS events for light-water-cooled nuclear power plants, 10 CFR 50.62(c)(4) requires each boiling water reactor (BWR) to have a SLC System with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLC System and its injection location must be designed to perform its function in a reliable manner. The SLC system initiation must be automatic and must be designed to perform its function in a reliable manner.

Each BWR is also required to have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device and equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS.

General Design Criteria (GDC) 26 requires that two independent reactivity control systems of different design principles shall be provided. However, only one of the systems is required to be capable of holding the reactor core subcritical under cold conditions. Cold conditions are OPERATIONAL CONDITION 4 (COLD SHUTDOWN) or OPERATIONAL CONDITION 5 (REFUELING). The control rod system is capable of holding the reactor core subcritical under cold conditions. The SLC System provides backup capability for reactivity control (i.e., achieving cold shutdown), independently of the normal reactivity control system. Therefore, GDC 26 is met.

GDC 27 requires the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. The implementation of this TS change does not have any impact on the ability of the reactivity control systems to meet the requirements of GDC 27.

The SLC system is only required under an extremely low probability event, where all of the control rods are assumed to be inoperable while the reactor is at normal full power operation. In this event, not enough control rods can be inserted into the reactor core to accomplish shutdown and cooldown in the normal manner. The reactor scram function of the Control Rod Drive System, backed up by the ARI

function, is expected to ensure prompt shutdown of the reactor when required. The operator control console and the emergency response information displays provide abnormal status information, including reactor power, low water level, and high dome pressure to indicate if SLC System initiation is needed.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

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

   Response: No.

   The proposed changes to delete the operability requirement for the SLC System in OPERATIONAL CONDITION 5* (OPERATIONAL CONDITION 5 with any control rod withdrawn) does not affect the probability or consequences of an accident previously evaluated. In STARTUP and POWER OPERATION, the SLC System is required to provide shutdown capability. In HOT SHUTDOWN and COLD SHUTDOWN, control rods are not able to be withdrawn since the reactor mode switch is in Shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. Design basis accident mitigation scenarios for OPERATIONAL CONDITION 5 do not depend on, or require, SLC System operability. In REFUELING mode, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate shutdown margin in accordance with TS LIMITING CONDITION FOR OPERATION 3.1.1 ensures that the reactor will not become critical. Since the purpose of the SLC System is to bring the reactor to a cold shutdown condition from normal power operations and maintain it in a cold shutdown condition, there is no design basis for the SLC System to be required to be OPERABLE when only a single control rod can be withdrawn. In addition, the reactor protection system and the control rod system would continue to be able to provide protection in the unlikely event that an inadvertent criticality occurs.

   Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Specifically, no new hardware is being added to the plant as part of the proposed change, no existing equipment is being modified, and no significant changes in operations are being introduced.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The proposed changes will not alter any assumptions, initial conditions, or results of any accident analyses. The purpose of the SLC System is to bring the reactor to and maintain it in a cold shutdown condition following a failure to scram during plant operations. The SLC System is not designed to terminate an inadvertent criticality during REFUELING. Shutdown margin, either demonstrated or analytically determined, in accordance with Technical Specifications and procedural controls, will assure that an inadvertent criticality event will not occur during REFUELING. In addition, the reactor protection system and control rod system provide protection in the unlikely event that an inadvertent criticality occurs. The proposed change does not affect the ability of the SLC System to achieve plant shutdown under analyzed conditions (POWER OPERATION and STARTUP).

   Therefore, this change does not involve a significant reduction in a margin of safety.

   Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.62 (c)(4) requires that each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner. In addition, as discussed above, GDC 26 and 27 must continue to be met.

The requirement for SLC to be OPERABLE during REFUELING was removed from Standard Technical Specifications because the SLC System is not required during refueling since only a single control rod can be withdrawn and adequate SHUTDOWN MARGIN prevents criticality under these conditions.

In conclusion, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL IMPACT EVALUATION

PSEG has determined the proposed amendment relates to changes in a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or relates to changes in an inspection or a surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore,

pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed change is not required.

7. REFERENCES

7.1 Code of Federal Regulations, 10 CFR 50.62.
7.2 NUREG –1433, "Standard Technical Specifications General Electric Plants, BWR/4."
7.3 Hope Creek, Updated Final Safety Analysis Report.
7.4 Hope Creek, Technical Specifications.
7.5 Issuance of Amendment- Nine Mile Point Nuclear Station, Unit 2, by letter dated September 30, 1993 (TAC No. M86186).
7.6 Issuance of Amendments- Susquehanna Steam Electric Station, Units 1 and 2, by letter dated December 20, 1994 (TAC Nos. M89666 & M89667)
7.7 Issuance of Amendments - Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, by letter dated March 6, 1998 (TAC Nos. M99907, M99908, M99943, and M99944)
7.8 Issuance of Amendment - Duane Arnold Energy Center by letter dated March 31, 1998 (TAC No. MA0780).

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Hope Creek Generating Station Facility Operating License NPF-57 are affected by this change request:

Technical Specification	Page
3.1.5.b	3/4 1-19
Table 3.3.2-1, item 4.e	3/4 3-12 and 3/4 3-16a
Table 4.3.2.1-1, item 4.e	3/4 3-29 and 3/4 3-31

REACTIVITY CONTROL SYSTEMS

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system consists of two redundant subsystems and shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and ~~5*~~

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one system subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With both system subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one system subsystem inoperable, restore subsystem to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

2. With both standby liquid control system subsystems inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

1. The temperature of the sodium pentaborate solution in the storage tank is greater than or equal to 70°F.

2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-1.

3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 70°F.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION		VALVE ACTUATION GROUPS OPERATED BY SIGNAL[d]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [a]	APPLICABLE OPERATIONAL CONDITION	ACTION
3.	MAIN STEAM LINE ISOLATION				
a.	Reactor Vessel Water Level - Low Low Low, Level 1	1	2	1, 2, 3	21
b.	Main Steam Line Radiation - High, High	2[b]	2	1, 2, 3##	28
c.	Main Steam Line Pressure - Low	1	2	1	22
d.	Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e.	Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f.	Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g.	Manual Initiation	1, 2, 17	2	1, 2, 3	25
4.	REACTOR WATER CLEANUP SYSTEM ISOLATION				
a.	RWCU Δ Flow - High	7	1/Valve[e]	1, 2, 3	23
b.	RWCU Δ Flow - High, Timer	7	1/Valve[e]	1, 2, 3	23
c.	RWCU Area Temperature - High	7	6/Valve[e]	1, 2, 3	23
d.	RWCU Area Ventilation Δ Temperature-High	7	6/Valve[e]	1, 2, 3	23
e.	SLCS Initiation	7[f]	1/Valve[e]	1, 2, 5#	23
f.	Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve[e]	1, 2, 3	23
g.	Manual Initiation	7	1/Valve[e]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

NOTES

* When handling recently irradiated fuel in the secondary containment and during operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

~~# Refer to Specification 3.1.5 for applicability.~~

The hydrogen water chemistry (HWC) system shall not be placed in service until reactor power reaches 20% of RATED THERMAL POWER. After reaching 20% of RATED THERMAL POWER, and prior to operating the HWC system, the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER and after the HWC system has been shutoff, the background level and associated setpoint shall be returned to the normal full power values. If a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also trips and isolates the mechanical vacuum pumps.

(c) Also starts the Filtration, Recirculation and Ventilation System (FRVS).

(d) Refer to Table 3.3.2-1 table notation for the listing of which valves in an actuation group are closed by a particular isolation signal. Refer to Tables 3.6.3-1 and 3.6.5.2-1 for the listings of all valves within an actuation group.

(e) Sensors arranged per valve group, not per trip system.

(f) Closes only RWCU system isolation valve(s) HV-FO01 and HV-F004.

(g) Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.

(h) Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.

(i) Manual isolation closes HV-FO03 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.

(j) Trip functions common to RPS instrumentation.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
MAIN STEAM LINE ISOLATION (Continued)				
e. Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
g. Manual Initiation	NA	Q(a)	NA	1, 2, 3
4. **REACTOR WATER CLEANUP SYSTEM ISOLATION**				
a. RWCU Δ Flow - High	S	Q	R	1, 2, 3
b. RWCU Δ Flow - High, Timer	NA	Q	R	1, 2, 3
c. RWCU Area Temperature - High	NA	Q	R	1, 2, 3
d. RWCU Area Ventilation Δ Temperature - High	NA	Q	R	1, 2, 3
e. SLCS Initiation	NA	Q(b)	NA	1, 2, 5#
f Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3
g. Manual Initiation	NA	Q(a)	NA	1, 2, 3
5. **REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION**				
a. RCIC Steam Line Δ Pressure (Flow) - High	NA	Q	R	1, 2, 3
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA	Q	R	1, 2, 3
c. RCIC Steam Supply Pressure - Low	NA	Q	R	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
h. HPCI Torus Compartment Temperature - High	NA	Q	R	1, 2, 3
i. Drywell Pressure - High	NA	Q	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	Q	R	1, 2, 3
c. Manual Initiation	NA	Q[(a)]	NA	1, 2, 3

* When handling recently irradiated fuel in the secondary containment and during operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

~~# Refer to Specification 3.1.5 for applicability.~~

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 92 days.

RETYPED TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Hope Creek Generating Station Facility Operating License NPF-57 are affected by this change request:

Technical Specification	Page
3.1.5.b	3/4 1-19
Table 3.3.2-1, item 4.e	3/4 3-12 and 3/4 3-16a
Table 4.3.2.1-1, item 4.e	3/4 3-29 and 3/4 3-31

3/4.1 5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system consists of two redundant subsystems and shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

   1. With one system subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

   2. With both system subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

   1. The temperature of the sodium pentaborate solution in the storage tank is greater than or equal to 70°F.

   2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-1.

   3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 70°F.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL[(d)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	1	2	1, 2, 3	21
b. Main Steam Line Radiation - High, High	2[(b)]	2	1, 2, 3##	28
c. Main Steam Line Pressure - Low	1	2	1	22
d. Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f. Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g. Manual Initiation	1, 2, 17	2	1, 2, 3	25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	7	1/Valve[(e)]	1, 2, 3	23
b. RWCU Δ Flow - High, Timer	7	1/Valve[(e)]	1, 2, 3	23
c. RWCU Area Temperature - High	7	6/Valve[(e)]	1, 2, 3	23
d. RWCU Area Ventilation Δ Temperature-High	7	6/Valve[(e)]	1, 2, 3	23
e. SLCS Initiation	7[(f)]	1/Valve[(e)]	1, 2	23
f. Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve[(e)]	1, 2, 3	23
g. Manual Initiation	7	1/Valve[(e)]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

NOTES

*	When handling recently irradiated fuel in the secondary containment and during operations with a potential for draining the reactor vessel.
**	When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.
##	The hydrogen water chemistry (HWC) system shall not be placed in service until reactor power reaches 20% of RATED THERMAL POWER. After reaching 20% of RATED THERMAL POWER, and prior to operating the HWC system, the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER and after the HWC system has been shutoff, the background level and associated setpoint shall be returned to the normal full power values. If a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.
(a)	A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b)	Also trips and isolates the mechanical vacuum pumps.
(c)	Also starts the Filtration, Recirculation and Ventilation System (FRVS).
(d)	Refer to Table 3.3.2-1 table notation for the listing of which valves in an actuation group are closed by a particular isolation signal. Refer to Tables 3.6.3-1 and 3.6.5.2-1 for the listings of all valves within an actuation group.
(e)	Sensors arranged per valve group, not per trip system.
(f)	Closes only RWCU system isolation valve(s) HV-FOO1 and HV-F004.
(g)	Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.
(h)	Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.
(i)	Manual isolation closes HV-FO03 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.
(j)	Trip functions common to RPS instrumentation.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION		CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
MAIN STEAM LINE ISOLATION (Continued)					
e.	Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f.	Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
g.	Manual Initiation	NA	Q(a)	NA	1, 2, 3
4.	REACTOR WATER CLEANUP SYSTEM ISOLATION				
a.	RWCU Δ Flow - High	S	Q	R	1, 2, 3
b.	RWCU Δ Flow - High, Timer	NA	Q	R	1, 2, 3
c.	RWCU Area Temperature - High	NA	Q	R	1, 2, 3
d.	RWCU Area Ventilation Δ Temperature - High	NA	Q	R	1, 2, 3
e.	SLCS Initiation	NA	Q(b)	NA	1, 2
f	Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3
g.	Manual Initiation	NA	Q(a)	NA	1, 2, 3
5.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a.	RCIC Steam Line Δ Pressure (Flow) - High	NA	Q	R	1, 2, 3
b.	RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA	Q	R	1, 2, 3
c.	RCIC Steam Supply Pressure - Low	NA	Q	R	1, 2, 3
d.	RCIC Turbine Exhaust Diaphragm Pressure - High	NA	Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
h. HPCI Torus Compartment Temperature - High	NA	Q	R	1, 2, 3
i. Drywell Pressure - High	NA	Q	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	Q	R	1, 2, 3
c. Manual Initiation	NA	Q[(a)]	NA	1, 2, 3

* When handling recently irradiated fuel in the secondary containment and during operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 92 days.

PROPOSED CHANGES TO TS BASES PAGES

The following Technical Specifications Bases for Hope Creek Generating Station, Facility Operating License No. NPF-57, is affected by this change request:

Technical Specification	Page
Bases 3/4.3.2	B 3/4 3-2k

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The valve groups actuated by this Function are listed in Table 3.3.2-1.

4.c, 4.d. RWCU Area Temperature and Area Ventilation Differential Temperature - High

RWCU area temperatures and area ventilation differential temperatures are provided to detect a leak from the RWCU System. The isolation occurs even when very small leaks have occurred and is diverse to the high differential flow instrumentation for the hot portions of the RWCU System. If the small leak continues without isolation, offsite dose limits may be reached. Credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

Area temperature and area ventilation differential temperature signals are initiated from temperature elements that are located in the room that is being monitored. Twelve ambient temperature sensor/monitors provide input to the RWCU Area Temperature - High Function. Twelve channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

Twelve differential temperature sensor/monitors provide input to the RWCU Area Ventilation Differential Temperature - High Function. Twelve channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The valve groups actuated by this Function are listed in Table 3.3.2-1.

4.e. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System. SLC System initiation signals are initiated from the two SLC pump start signals.

Two channels (one from each pump) of the SLC System Initiation Function are available and are required to be OPERABLE only in OPERATIONAL CONDITIONS 1 & 2 ~~and 3~~ (when the SLC system is required to be OPERABLE), since these OPERATIONAL CONDITIONS are consistent with the Applicability for the SLC System (TS 3.1.5).

4.f. Reactor Vessel Water Level - Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level - Low Low, Level 2 Function associated with RWCU isolation is not